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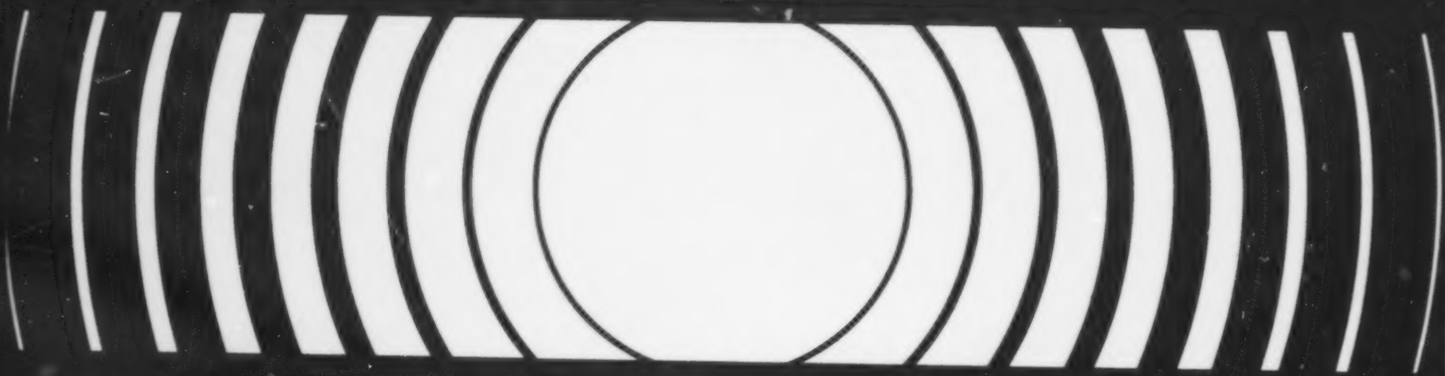
Radiological Impact Caused by Emissions of Radionuclides into Air in the United States

Preliminary Report



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Radiological Impact Caused by Emissions of Radionuclides into Air in the United States

Preliminary Report

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SUMMARY

Radionuclides are emitted into the air from numerous sources located throughout the United States, including such facilities as nuclear power plants and other facilities pertaining to the nuclear fuel cycle, national defense facilities, industrial plants, research and development laboratories (including research reactors and accelerators), medical facilities, certain mining and milling operations, and fossil fuel combustion plants. As a result of the operation of these facilities, radionuclides are released into the atmosphere where they are dispersed into populated areas. Radiation exposure to the public can then occur by breathing or swallowing these materials. For some radionuclides, people can also be exposed from direct radiation that is emitted from a cloud of the material passing overhead or from direct radiation emitted when the radionuclides settle onto the ground.

Section 122 of the Clean Air Act Amendments of 1977, Public Law 95-95, directed the Administrator of the Environmental Protection Agency, to review all relevant information and determine whether emissions of radioactive pollutants into ambient air will cause or contribute to air pollution which may reasonably be anticipated to endanger public health.

As a part of this review, the Agency has been assessing the public health impact resulting from emissions of radionuclides into air from a broad spectrum of major source categories. This report presents the initial, preliminary results of these assessments. For each facility or source category, the following are presented:

- the amount of radionuclides released into the atmosphere;
- the radiation doses to individuals and population groups;
- the lifetime risks to individuals;
- the number of fatal cancers in the exposed population per year of facility operation.

These preliminary results were obtained by using the available data base on emission levels and the current calculational methods for estimating dose and health risks. Simultaneously with the development of this report, the Agency has been conducting more in-depth assessments which are expected to be published within the next six months. These assessments will be based on further evaluations of emission levels, including the use of results from recent field measurement studies and computer programs which will incorporate more recently developed dose and health risk information and methodology.

Source categories in this report have been divided into four groups:

- facilities licensed by the Nuclear Regulatory Commission or States under an agreement with the Nuclear Regulatory Commission;
- facilities operated and regulated by the Department of Energy;
- facilities emitting naturally occurring radionuclides;
- other facilities emitting minor amounts of radionuclides.

Summaries of the emissions, dose rates, and risks associated with representative model facilities of source categories within these groups and with actual Department of Energy facilities are shown in tables S-1 through S-4. These data should be treated as preliminary estimates and used carefully with the recognition that they are highly uncertain.

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Table S-1. Summary of radiological impact caused by atmospheric emissions from licensed[a] sources

Source category	Number of sources	Model facility						
		Principal radionuclide emissions (Ci/y)		Principal dose equivalent rates: Maximum individual (mrem/y)[b]		Principal dose equivalent rates: Regional Population (Person-rem/y)[c]	Lifetime risk to the maximum individual (x 10^{-6})	Expected fatal cancers per year of operation: Regional population (Fatal cancers)
Uranium conversion facilities (2.1A)	2	U-238 U-234 Th-234	0.08 0.08 0.08	Lung Bone	88 9	120 5	300	0.007
Fuel fabrication facility (2.1B)	16	U-234 U-235 U-236 U-238 Th-231 Th-234	0.008 0.0003 0.0004 0.001 0.0003 0.001	Lung Bone	5 0.5	7 0.3	20	0.0004
Light water reactors								
Boiling water reactors (2.1C)	25	Noble gases Tritium Halogens	7,000 43 1.4	Thyroid Whole body	11 2	19 8	20	[d]0.001
Pressurized water reactors (2.1C)	44	Noble gases Tritium Halogens	13,000 1,100 0.05	Thyroid Whole body	1 0.8	7 5	9	[d]0.001
High temperature gas cooled reactor (Fort St. Vrain) (2.2)	1	Noble gases	1,100	Whole body	0.5	6	6	0.001
Radiopharmaceutical industry Producers (2.3)	34	I-125 I-131	0.9 0.9	Thyroid Whole body	86 0.2	200 0.7	8	0.0003

See footnotes at end of table.

Table S-1. Summary of radiological impact caused by atmospheric emissions from licensed[a] sources--continued

Source category	Model facility							
	Number of sources	Principal radionuclide emissions (Ci/y)		Principal dose equivalent rates			Lifetime risk to the maximum individual ($\times 10^{-6}$)	Expected fatal cancers per year of operation
				Maximum individual (mrem/y)[b]		Regional Population (Person-rem/y)[c]		Regional population (Fatal cancers)
Test reactors (2.4)	2	Ar-41 Tritium	450 150	Whole body	0.2	1	2	0.0002
Research reactors (2.5)	13	Ar-41 Tritium	200 100	Whole body	0.09	0.5	1	0.00008
University reactors (2.6)	54	Ar-41 Tritium	1,000 100	Whole body	0.4	2	5	0.0004
Shallow land burial Low level waste (2.7)	6	Tritium	19,000	Bone Whole body	30 20	6 5	90	0.0008
Plutonium Fabrication Facilities (2.8)	7	Plutonium Americium	0.00006 0.000004	Bone	0.4	0.6	1	<0.00001
Industrial users (2.9)	4,000	Mostly sealed	No data					

[a] Sources licensed by the Nuclear Regulatory Commission (NRC) or States which have entered into an agreement with the NRC whereby certain regulatory authority is relinquished by the NRC and assumed by the States pursuant to Section 274 of the Atomic Energy Act of 1954, as amended.

[b] The maximum dose equivalent rate an individual is likely to receive living near the facility.

[c] The maximum collective dose equivalent rate to the regional population. This is the maximum value expected to occur within 100 years following the start of facility operation.

[d] The number of fatal cancers to the worldwide population has been estimated to be 0.04 from BWRs and 0.02 from PWRs per year (See Section 2.1C).

Table S-2. Summary of radiological impact caused by atmospheric emissions from Department of Energy facilities

Facility	Number of facilities	Principal radionuclide emissions (Ci/y)		Principal dose equivalent rates[a] Maximum individual (mrem/y)		Principal dose equivalent rates[a] Regional Population (Person-rem/y)	Lifetime risk to the maximum individual (x 10^{-6})		Expected fatal cancers per year of operation Regional population (Fatal cancers)
Savannah River Plant[b] (3.2)	1	Ar-41 C-14 Tritium Kr-85	65,000 63 360,000 440,000	Total body	1.1	110	Site boundary	20	0.02
Los Alamos Scientific Laboratory (3.4)	1	Ar-41 C-11 Tritium N-13 O-15	800 14,000 39,000 1,400 32,000	Total body	70	11	Site boundary Nearest resident	1,000 300	0.002
Lawrence Livermore Laboratory (3.5)	1	Ar-41 Tritium N-13 O-15	380 5,200 590 390	Total body	4.3	2.9	Site boundary Nearest resident	60 10	0.0006
Rocky Flats Plant (3.6)	1	Pu-239 & Pu-240 U-235 U-238	0.000004 0.00002 0.00002	Lung Bone Kidney Liver	0.08 0.035 0.004 0.005	130 58 7 9	Site boundary	0.3	0.007
Argonne National Laboratory (3.12)	1	Ar-41 Tritium	30,000 850	Total body	5.6	180	Site boundary Nearest resident	80 40	0.04
Brookhaven National Laboratory (3.13)	1	Ar-41 Tritium O-15	360 1,200 67,000	Total body	0.16	19	Site boundary	2	0.004

See footnotes at end of table.

Table S-2. Summary of radiological impact caused by atmospheric emissions from Department of Energy facilities--continued

Facility	Number of facilities	Principal radionuclide emissions (Ci/y)		Principal dose equivalent rates[a]		Lifetime risk to the maximum individual ($\times 10^{-6}$)		Expected fatal cancers per year of operation
				Maximum individual (mrem/y)	Regional Population (Person-rem/y)			Regional population (Fatal cancers)
Oak Ridge Facilities[c] (3.14)	1	U-234 I-131 Kr-85	0.06 1.4 8,600	Lung 4.6 Bone 5.5 Thyroid 1.5	Total body 4.7	Nearest resident	10	0.001
Portsmouth Gaseous Diffusion Plant[c] (3.15)	1	Th-234 U-234 Tc-99	0.05 0.1 4.5	Lung 1.1 Bone 1.2 Kidney 1.1 G. I. Tract 2.2	Total body 0.19	Site boundary	10	0.00004
Paducah Gaseous Diffusion Plant[c] (3.16)	1	U-234 U-238	0.2 0.2	Lung 6.3 Bone 0.9	NR	Site boundary	20	NR
Ames Laboratory (3.17)	1	Ar-41 Tritium	13,000 1,000	Total body 5.5	250	Site boundary	80	0.05
Other sites	17	Various	Small	Total body<1	Total body <10	Site boundary	<10	<0.002

[a] Data taken directly from DOE reports. NR Not reported.
[b] All doses are 70-year dose commitments.
[c] All doses are 50-year dose commitments.

Table S-3. Summary of radiological impact caused by atmospheric emissions of natural radioactive materials

Source category	Number of sources	Model facility: Principal radionuclide emissions (Ci/y)[a]		Exposure levels		Principal dose equivalent rates			Lifetime risk to the maximum individual (x 10^{-6})	Expected fatal cancers per year of operation: Population (Fatal cancers)	
				Maximum individual (WL)	Regional Population (Person-WL)		Maximum Individual (mrem/y)[b]	Regional Population (Person-rem/y)[c]		Regional	U.S.
Uranium Mines (4.1)											
Underground	251	Rn-222	6700	0.006	1.3		-	-	10,000	0.03	0.08
Open pit	36	Rn-222	2000	0.0008	0.4		-	-	1,000	0.008	0.02
Uranium Mills (4.2)	20	Rn-222 U-238+d	2700 0.4	0.005	0.5	Lung Bone	350 360	3.4 3.9	10,000	0.01	0.03
Phosphate Industry (4.3)											
Mining and beneficiation	35	Rn-222	1300	0.0002	4.9		-	-	300	0.1	
Drying and grinding facilities	20	Rn-222 U-238+d	20 0.03	0.00005	0.08	Lung Bone	54 79	17 18	500	0.004	
Phosphoric acid plant	35	Rn-222 U-238+d	480 0.1	0.0007	2.0	Lung Bone	85 110	46 45	2,000	0.05	
Elemental phosphorus plant	9	Rn-222 U-238+d Po-210	490 0.15 7.4	0.0004	2.0	Lung Bone Kidney	740 570 1800	770 440 1400	6,000	0.1	
Coal-fired power stations[d] (4.4)											
New stations	145	Rn-222 U-238+d Th-232+d	1.9 0.3 0.07	<0.00001	<0.00001-0.024	Lung Bone	0.8-2.1 1.6-16	0.5-2600 1.5-1300	10-60	0.00008-0.2	
Existing stations	250	Rn-222 U-238+d Th-232+d	0.7 0.8 0.3	<0.00001	<0.00001-0.013	Lung Bone	6.7-15 8.3-62	2.7-19,000 7.1-11,000	60-700	0.0004-1.5	

See footnotes at end of table.

Table S-3. Summary of radiological impact caused by atmospheric emissions of natural radioactive materials--continued

		Model facility								
				Exposure levels		Principal dose equivalent rates			Expected fatal cancers per year of operation	
				Maximum	Regional	Maximum	Regional	Lifetime risk to the maximum	Population	
Source category	Number of sources	Principal radionuclide emissions (Ci/y)[a]		individual (WL)	Population (Person-WL)	Individual (mrem/y)[b]	Population (Person-rem/y)[c]	individual (x 10^{-6})	Regional (Fatal cancers)	U.S.
Metal mining and milling[e] (4.5)	177	Rn-222	17 to 3000	<0.00001 to 0.001	0.005 to 0.8	-	-	20 to 2000	0.0001 to 0.02	
Nonmetal mining and milling[f] (4.5)	4,200	Rn-222	0.2 to 18	<0.00001 to 0.00004	0.0007 to 0.06	-	-	0.7 to 50	0.00002 to 0.001	
Geothermal power site (4.6A)	1	Rn-222	540	0.0004	1.5	-	-	700	0.03	
Ground water treatment plants (4.6B)	40,000	Rn-222	3.4	<0.00001	0.06	-	-	10	0.001	

[a] U-238+d emission rates are the sum of the individual release rates for uranium-238, and its daughter products uranium-234, thorium-230, radium-226, lead-210, and polonium-210. (For elemental phosphorus plants only, polonium-210 is not included in this sum.)

Th-232+d emission rates are the sum of the individual release rates for thorium-232 and its daughter products radium-228, thorium-228, and radium-224.

[b] The maximum dose equivalent rate an individual is likely to receive living near the facility.

[c] The maximum collective dose equivalent rate to the regional population. This is the maximum value expected to occur within 100 years following the start of facility operation.

[d] Ranges of impact values represent variations due to station siting.

[e] Includes iron, copper, zinc, and bauxite.

[f] Includes clay, limestone, fluorspar.

Table S-4. Summary of radiological impact caused by atmospheric emissions from Department of Defense facilities

		Model facility					
				Principal dose equivalent rates			Expected fatal cancers per year of operation
Source category	Number of sources	Principal radionuclide emissions (Ci/y)		Maximum individual (mrem/y)^b	Regional Population (Person-rem/y)^c	Lifetime risk to the maximum individual (x 10^{-6})	Regional population (Fatal cancers)
Army (5.1A)	2	Argon-41	12	-	-	-	No off-site doses reported
Navy (5.1B) (Shipyard operations)	9	Argon-41 Cobalt-60 Tritium Carbon-14 Noble gases	0.4 0.003 0.001 0.1 0.6	Total body <2	<1	<30	<0.0002
Accelerators (5.2)	1200	Tritium Carbon-11 Nitrogen-13 Oxygen-15 Argon-41	1 0.002 0.04 1.0 0.0001	Whole body <1	<1	<10	<0.0001

CHAPTER 1

INTRODUCTION

1.1 Content of Report

This report presents a preliminary analysis of the public health impact caused by emissions of radionuclides into air from major source categories in the United States. For each source category, the following information is presented:

(1) A general description of the source category and its size.

(2) A brief description of the processes or activities which lead to the emission of radioactive materials into the air.

(3) A summary of emissions data for the source category.

(4) Estimates of the radiation doses and health risks to both individuals and population groups.

1.2 Source Categories

Radionuclides are emitted into the air from numerous sources located throughout the United States including nuclear power plants and other facilities in the nuclear fuel cycle, national defense facilities, research and development laboratories (including research reactors and accelerators), medical facilities, industrial users, certain mining and milling operations, and fossil fuel combustion plants.

For the purposes of the assessments presented in this report, these sources have been divided into four major categories: (1) facilities licensed by the Nuclear Regulatory Commission (NRC) or States* under an agreement with NRC; (2) facilities operated and regulated under the direction of the Department of Energy (DOE); (3) facilities emitting naturally occurring radioactive materials; and (4) other minor sources of emissions.

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*Sources are licensed by the Nuclear Regulatory Commission (NRC) or States which have entered into an agreement with the NRC whereby certain regulatory authority is relinquished by the NRC and assumed by the States pursuant to Section 274 of the Atomic Energy Act of 1954, as amended.

DOE facilities were analyzed as single sources on a site-by-site basis. The NRC and State licensed facilities, DOD facilities, and the facilities emitting naturally occurring radionuclides were grouped together into source categories on the basis of similarity of activities or operations and analyzed on a generic basis.

1.3 Emissions Data

An attempt was made to gather and document available information on the radioactive emissions from the various activities which use radionuclides and to use these data to estimate the public health impact resulting from these emissions.

The availability of emissions data varied widely from very detailed data for some source categories to little or no information for other source categories. Where possible the analyses presented are based on the use of reported emissions data obtained by measurement. However, in the absence of measurement data, best estimates based on calculated or extrapolated values were used. The present status of information on emissions for the various source categories and the origin of the data used in the health impact analyses are discussed under the subsections entitled "Emissions of Radionuclides."

1.4 Health Impact Assessment

The data on radioactive emissions has been used to estimate the public health impact of these emissions. These assessments include estimates of the following radiation exposures and health risks:

(1) Dose-equivalent rates and working level exposures to the most exposed individuals (maximum individual);

(2) Collective dose-equivalent rates and working level exposures to population groups;

(3) Lifetime risks to the maximum and average individuals in the exposed population;

(4) The number of fatal cancers committed in the exposed population per year of facility operation.

The health risks estimated in this report are for fatal cancers only. Our current practice is to assume that for whole body exposure, the number of genetic health effects and the number of nonfatal cancers are each about the same as the number of fatal cancers (EPA77).

Assessment Methodology

The dose assessments for the DOE facilities sites were carried out on a site-by-site basis using data directly from DOE reports (Chapters 3). DOD facilities were assessed using site specific and generic assessments from DOD reports (Chapter 5).

The dose assessments for the NRC and State licensed facilities, and for sources emitting naturally occurring radioactive materials (Chapters 2 and 4) were carried out on a generic basis using model facilities located on generic sites. These generic dose assessments were carried out using the AIRDOS-II (Mo77) computer code with some minor modifications. This methodology is described in detail in Appendix A. Cancer risks have been estimated using the risk per rem and risk per working level-year* level conversion factors shown in table B-1 of Appendix B. The risk/rem conversion factors were developed from information in the BEIR report (BEIR72). The risk/WL-y conversion factors were developed primarily from information on uranium miner exposures (EPA79 Section 4.0).

Dose and health risk methodology has evolved substantially in recent years. Some of these changes have not yet been incorporated in the methodology used in carrying out the assessments presented in this report. However, because of the preliminary nature of this report and the limited time available for its preparation, the use of the existing (off the shelf) methodology and DOE and DOD data were believed to be statisfactory for this effort.

Future health impact assessments of radioactive emissions under the Clean Air Act will use more recently developed methodology. The Oak Ridge National Laboratory (ORNL) under contract to EPA is developing computer programs incorporating this revised methodology. The existing calculational models are being revised based on the latest information on transport, uptake and metabolic behavior of the various radionuclides. In particular, these programs will use ORNL's INREM-II and S factor codes. The cancer risk methodology will use "life-table" data which considers age-specific characteristics of somatic health effects. The life-table analysis will consider relative and or absolute risk for each radionuclide and latency periods and risks plateaus for each type of health effect where possible. Information from the BEIR-III report (BEIR79) will be incorporated into this methodology following publication of this report.

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*A person exposed to one working level continuously for one year is considered to have received an integrated exposure equal to one person-working level year.

Definition of Terms Used in Health Impact Assessment

Maximum Individual

Dose equivalent rates and working level exposures are presented for what is titled the "maximum individual." For DOE and DOD sites the maximum individual represents either a hypothetical individual at the site boundary or the nearest actual individual or group of individuals. For NRC and State licensed facilities and sources of natural radioactive materials, the maximum individual represents those individuals living closest to the source of emissions. For the generic assessments, these individuals were considered to be located approximately 500 meters from the point of release in the predominant wind direction. For area sources this location was nominally 500 meters from the edge of the source. In the cases of elevated releases where the location of maximum exposure was at a distance beyond 500 meters, the dose to the maximum individual was calculated at the location of maximum exposure.

The dose rates presented for the maximum individual are 50-year committed dose equivalents. This is the dose equivalent that will be accumulated over a 50-year period following an intake.

The working level exposures presented for the maximum individual are the radon-222 decay product levels to which an individual would be exposed assuming 70 percent equilibrium (i.e., 100 p Ci/L radon-222 = 0.7 WL).

Average Individual

Dose rates and working level exposures for an average individual within 80 km of a source were obtained by dividing the collective dose rates and working level exposures for the region by the population of the region.

Population

The term population refers to the population living within a radius of 80 kilometers of a source unless otherwise noted in the text. For a few source categories, exposures are presented for the population of the United States or the World and these cases are specifically identified in the appropriate tables.

Collective dose equivalent rates and working level exposures are expressed in units of person-rem/year and person-working levels and are the sum of the dose equivalent rates or working level exposures to all the individuals in the exposed population due to the releases from a source. Further details of these calculations are contained in Appendix A.

Individual Lifetime Risks and Number of Fatal Cancers

The individual lifetime risks are the fatal cancer risks to individuals which would result from a lifetime of exposure (70 years) to the doses and working levels estimated for those individuals. The lifetime risk to the maximum individual was obtained by multiplying the dose equivalent rates and working level exposures by 70 to obtain the lifetime exposure and then multiplying this value by risk/rem or risk/WL-year factors as described Appendix B.

The lifetime risk to the average individual was obtained by dividing the population exposures in person-rem/year and person-working levels by the total number of people in the exposed population to obtain average annual exposures rates and then proceeding as described above for the maximum individual.

The number of fatal cancers per year were obtained by multiplying the annual collective dose equivalents and working level exposures by risk/rem or risk/WL-year factors as described in Appendix B.

Numeric notation and units

Throughout this report, numeric values are frequently expressed in a modified scientific format. For example, 0.00123 = 1.23×10^{-3} may be expressed as 1.23E-3 and 3210 = 3.21×10^{3} as 3.21E+3.

Metric units have been used for reporting data, except in a few instances where referenced data are presented in the original units.

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CHAPTER 2
SOURCES OF EMISSIONS FROM FACILITIES LICENSED BY THE NUCLEAR REGULATORY COMMISSION

2.1. Uranium Fuel Cycle

The Environmental Radiation Protection Standards for Nuclear Power Operations published by EPA defined the uranium fuel cyle as "... the operations of milling of uranium ore, chemical conversion of uranium, isotopic enrichment of uranium, fabrication of uranium fuel, generation of electricity by a light-water-cooled nuclear power plant using uranium fuel, and reprocessing of spent uranium fuel, to the extent that these directly support the production of electric power for public use utilizing nuclear energy, but excludes mining operations, operations at waste disposal sites, transportation of any radioactive material in support of these operations, and the reuse of recovered non-uranium special nuclear and by-product materials from the cycle" (EPA77). These standards (40 CFR 190) pertain to EPA responsibilities for the protection of the environment.

The uranium fuel cycle facilities which are operated as a part of the commercial nuclear fuel cycle are regulated by the Nuclear Regulatory Commission except for uranium mills located in the Agreement States of New Mexico, Colorado, Washington, and Texas. Enrichment facilities are operated and controlled by the Department of Energy.

Uranium mines and mills are discussed in sections 4.1 and 4.2 of this report. There are no commercial reprocessing plants operating in the United States today.

REFERENCES

EPA77 Environmental Protection Agency, 1977, Radiation Protection Programs, Environmental Radiation Protection Standards for Nuclear Power Operations, Federal Register, Vol. 42, No. 9, Thursday, January 13, 1977.

40CFR190 Code of Federal Regulations, Part 40, Protection of the Environment, Government Printing Office, Washington, D.C.

2.1A Uranium Conversion Facilities

2.1A.1 General Description

The conversion facility purifies and converts uranium oxide (U_3O_8) to uranium hexafluoride (UF_6), the chemical form in which uranium enters the enrichment plants. There are two existing commercial UF_6 production facilities (Kerr-McGee Corporation at Sequoyah, Oklahoma, and Allied Chemical Corporation at Metropolis, Illinois) with a combined capacity of about 17,000 MT of uranium per year in the form of UF_6 (table 2.1A-1). NRC (NRC76) has estimated that five new facilities will be needed by the year 2000 to meet fuel cycle requirements if neither uranium nor plutonium is recycled.

2.1A.2 Process Description

Two industrial processes are used for uranium hexafluoride production (EPA73), the dry hydrofluor method and the solvent extraction method. Each method is used to produce roughly equal quantities of uranium hexafluoride feed for enrichment; however, the radioactive effluents from the two processes differ substantially. The hydrofluor method releases radioactivity primarily in the gaseous and solid states, while the solvent extraction method releases most of its radioactive wastes dissolved in liquid effluents.

Dry Hydroflour Process

The hydrofluor process consists of reduction, hydrofluorination and fluorination of the ore concentrates to produce crude uranium hexafluoride, followed by fractional distillation to obtain a pure product. The dry hydrofluor process separates impurities either as volatile compounds or as solid constituents of ash.

Solvent Extraction Process

The solvent extraction process employs a wet chemical solvent extraction step at the start of the process to prepare high purity uranium for the subsequent reduction, hydrofluorination, and fluorination steps. The wet solvent extraction method separates impurities by extracting the uranium into organic solvent leaving the impurities dissolved in an aqueous solution.

Table 2.1A-1. Location and capacity of uranium conversion facilities in the United States (AEC74)

State and Company	Location	Capacity uranium/year (metric tons)
Illinois		
Allied Chemical Corp.	Metropolis	12,600
Oklahoma		
Kerr-McGee Nuclear Corp.	Sequoyah	4,500

2.1A.3 Emissions of Radionuclides

Because no irradiated material is handled by conversion facilities, all radionuclides present also occur in nature. These nuclides are radium, thorium, uranium, and their respective decay products. Uranium is the major source of radioactivity in the gaseous effluents. Possible chemical species of uranium effluents include U_3O_8, UO_2, UF_4, UF_6, $(NH_4)_2U_2O_7$, and UO_2F_2. In the wet solvent extraction method, uranium is present as uranyl nitrate which may also appear in gaseous effluents. Thus, the uranium may be released as both soluble and insoluble aerosols. The discharge to the environment is through low stacks and vents.

2.1A.4 Model Facility

In order to estimate population and individual radiation doses, a model facility (table 2.1A-2) was developed. The ORNL Case 2 model UF_6 plant with low impurity feed (Se77) was chosen for this analysis. Case 2 control technology represents the practical limits of technology which are readily available today.

The model facility employs the dry hydrofluor process which releases radioactivity primarily in the gaseous and solid state form. The choice of one model facility rather than two simplifies

Table 2.1A-2. Model uranium conversion facility

Parameter	Value
Type:	Fluorination-fractionation (dry hydrofluor) UF_6 plant
Ore grade:	Low impurity plant feed containing 2800 pCi of thorium-230 and 200 pCi of radium-226 per gram of natural uranium.
Annual capacity:	10,000 metric tons of uranium
Emission control:	Primary treatment, secondary bag filters on dust control streams and secondary or tertiary scrubbers on process off-gas streams
Stack:	
Height	10 meters
Plume rise	0.0

the analysis, and the selection of the dry hydrofluor process results in an analysis of the facility process which releases the largest amount of radioactive material to the atmosphere.

Model Facility Emissions

The atmospheric emissions of radionuclides from the model conversion facility are presented in table 2.1A-3. The source terms were taken from the ORNL Case 2 model UF_6 plant with low impurity feed (Se77) and were based on operating data where these data were available. Where data were not available, assumptions were chosen which tend to make the source terms slightly high. About two-thirds of the total airborne losses occur via the untreated building ventilation effluent. Conservative assumptions which tended to maximize radon releases were used in estimating radon source terms. For example, a radon emanation coefficient of one was assumed for yellow cake (Se77).

Table 2.1A-3. Atmospheric emissions of radionuclides from a model uranium conversion facility

Radionuclide	Emissions (Ci/y)
Uranium-238	8.3E-2
Uranium-234	8.3E-2
Uranium-235	2.0E-3
Thorium-234	8.2E-2
Protactinium-234m	8.2E-2
Thorium-230	9.3E-4
Radium-226	6.7E-5
Radon-222	9.2

2.1A.5 Health Impact Assessment of a Model Uranium Conversion Facility

Annual radiation doses and working level exposures resulting from atmospheric radioactive emissions from a model uranium conversion facility are presented in tables 2.1A-4 and 2.1A-5. These estimates are for a site located in the suburbs of a large Midwestern city (Site B, Appendix A).

The maximum individual dose equivalent rate occurred 503 meters downwind. The uranium was assumed to be released as insoluble particulates of uranium oxide and fluoride compounds. The lung dose equivalent rate to the maximum exposed individual is 88 mrem/yr.

The maximum individual dose equivalent rate for an actual facility could be lower than that calculated for the model facility. Consideration is given to additional factors which could result in a dose reduction, e.g., higher stack height, fraction of released uranium which is soluble, smaller amounts of home grown food by the maximum individual, maximum individual located farther than 503 meters downwind, etc. Further airborne dose reductions using additional control technology are possible. Two-thirds of the airborne releases from the model facility are from the building ventilation effluent. This effluent could be treated, but this would be expensive because of the large volume of air that must be handled.

Individual fatal cancer risks and committed population fatal cancers are presented in table 2.1A-6. The lifetime fatal cancer risk to the maximum exposed group of individuals is estimated to be approximately 2.8E-4. The majority of the cancers are fatal lung cancers. The lifetime fatal cancer risk of the average individual living within 80 kilometers of the model facility is estimated to be 2.1E-7. The number of fatal cancers per year of facility operation is estimated to be 7.3E-3 to the population living in the region around the facility.

Table 2.1A-4. Annual radiation doses due to atmospheric radioactive emissions from a model uranium conversion facility

Organ	Maximum individual (mrem/y)	Average individual (mrem/y)	Population (person-rem/y)
Lung	8.8E+1	5.0E-2	1.2E+2
Bone	8.5	2.1E-3	5.1
Kidney	2.1	5.8E-4	1.5
Liver	7.6E-1	3.0E-4	7.2E-1
Thyroid	9.2E-1	4.1E-4	1.0
G.I. tract	1.2	2.6E-4	6.5E-1
Other soft tissue	9.4E-1	4.2E-4	1.1

Table 2.1A-5. Working level exposures from radon-222 emissions from a model uranium conversion facility

Source	Maximum individual (WL)	Regional population (person-WL)
Model facility	9.8E-6	9.7E-2

Table 2.1A-6. Individual lifetime risks and number of fatal cancers due to radioactive airborne emissions from a model uranium conversion facility

Source	Individual lifetime risks Maximum individual	 Average individual	Expected fatal cancers per year of operation (Fatal cancers)
Particulates	2.7E-4	1.5E-7	5.3E-3
Radon-222	1.5E-5	5.8E-8	2.0E-3
Total	2.8E-4	2.1E-7	7.3E-3

REFERENCES

AEC74 U.S. Atomic Energy Commission, 1974, Environmental Survey of the Uranium Fuel Cycle, WASH-1248 (Directorate of Licensing, Fuels and Materials, U.S. Atomic Energy Commission, Washington, DC).

EPA73 Office of Radiation Programs, 1973, Environmental Analysis of the Uranium Fuel Cycle - Part I - Fuel Supply, EPA-520/9-73-003-C (U.S. Environmental Protection Agency, Office of Radiation Programs, Washington, DC).

EPA77 U.S. Environmental Protection Agency, 1977, Subchapter F-Radiation Protection Programs, Part 190 - Environmental Radiation Protection Standards for Nuclear Power Operations Federal Register, Vol. 42, No. 9, Thursday, January 13, 1977.

NRC76 U.S. Nuclear Regulatory Commission, 1976, Final Generic Environmental Statement on the Use of Recycle Plutonium in Mixed Oxide Fuel in Light Water Cooled Reactors, NUREG-0002, Vol. 3 (National Technical Information Service Springfield, VA).

SE77 Sears M.B., Blanco R.E., Finney B.C., Hill G.S., Moore R.E., and Witherspoon J.P. 1977, Correlation of Radioactive Waste Treatment Costs and the Environmental Impact of Waste Effluents in the Nuclear Fuel Cycle--Conversion of Yellowcake to Uranium Hexafluoride, Part I. The Fluorination-Fractionation Process, ORNL/NUREG/TM-7, Nuclear Regulatory Commission.

2.1B Fuel Fabrication

2.1B.1 General Description

LWR fuels are fabricated from uranium that has been enriched in its content of uranium-235. The enriching process occurs at a gaseous diffusion plant where the natural uranium is processed to increase the uranium-235 content from 0.7 up to 2 to 4 percent by weight. The uranium hexafluoride product is shipped to LWR fuel fabrication plants where it is converted to solid uranium dioxide pellets and inserted into zirconium tubes. The tubes are fabricated into fuel assemblies which are shipped to nuclear power plants (Pe75).

The NRC currently licenses 16 separate facilities to conduct fuel, processing and fabrication operations (table 2.1B-1).

Table 2.1B-1. NRC-licensed uranium fuel processing and fabrication facilities (NRC79)

Company	Location
Atomics International	Canoga Park, California
Babcock & Wilcox	Lynchburg, Virginia
Babcock & Wilcox	Apollo, Pennsylvania
Combustion Engineering	Hematite, Missouri
Combustion Engineering	Windsor, Connecticut
Exxon Nuclear	Richland, Washington
General Atomic	San Diego, California
General Electric	San Jose, California[a]
General Electric	Wilmington, North Carolina
Kerr-McGee	Crescent City, Oklahoma
Nuclear Fuel Services	Erwin, Tennessee
Texas Instruments	Attleboro, Massachusetts
U.S. Nuclear	Oak Ridge, Tennessee
United Nuclear	Montville, Connecticut
United Nuclear	Wood River, Rhode Island
Westinghouse Electric	Columbia, South Carolina[b]

[a] Facility now reduced to R & D activities on LWR fuel.

[b] Also authorized to receive and store sealed mixed oxide fuel rods and assemble mixed-oxide rods into fuel bundles.

2.1B.2 Process Description

The processing technology used for fuel fabrication can be divided into three basic operations: chemical conversion of UF_6 to UO_2; mechanical processing including pellet production and fuel element fabrication; and recovery of uranium from scrap and off-specification material. These operations are usually identical in the various facilities. The exception is the chemical conversion of UF_6 to UO_2 which may be accomplished by the ADU (ammonium diuranate) process or the DC (direct conversion) process. The ADU process converts UF_6 to $(NH_4)_2U_2O_7$ which is then calcined to UO_2 powder. The UF_6 which is received from the enrichment facility is vaporized and transferred to the reaction vessels. The UF_6 is hydrolyzed with water and neutralized with NH_4OH at a pH of 8 to 9 to form a slurry of ADU in an aqueous solution of ammonium fluoride and ammonium hydroxide. The ADU is recovered in a centrifuge and a clarifier and is subsequently dried and calcined to form UO_2 powder.

The DC process hydrolyzes the UF_6 and reduces the uranium directly to UO_2. Cylinders of UF_6 are placed in steam-heated cabinets to vaporize the contained UF_6. The UF_6 gas enters into a bed of UO_2F_2 particles which is fluidized by steam. The gas reacts with the steam on the hot, wet surface of the particles to form a coating of UO_2F_2. The reaction is:

$$UF_6 + 2H_2O \rightarrow UO_2F_2 + 4HF.$$

The particles of UO_2F_2 overflow to a product hopper at which point the particles are approximately 120 µm in diameter. After a given amount is accumulated, the batch is transferred to the next vessel where the bed is fluidized by steam and ammonia. A second reaction follows:

$$UO_2F_2+H_2 \rightarrow UO_2+HF.$$

A high percentage of the UO_2F_2 is converted to UO_2 in the second reactor, but the product goes into a third reactor where, by the same process, the reaction is carried to completion.

The gaseous effluent from each of the three converter vessels passes through a sintered nickel filter in the top of each vessel before going to the gaseous effluent treatment system where HF and particulates are removed from the off-gas stream. The most significant potential environmental impact results from the release of UF_6 and UO_2F_2 during the UF_2 to UO_2 conversion and during chemical operations in scrap recovery.

Control Technology

Airborne emission control technology differs for ADU and DC type facilities. Treated air streams for both types consist of process and ventilation components. Control technology presented here is taken from Pechin as represented by Case 2 ADU and DC facilities which were used in determining model facility source terms (Pe75).

In the ADU facility, process gas passes through liquid (water) scrubbers (90 percent removal of entrained solids) and HEPA filters (95 percent efficient) before release to the atmosphere. Ventilation off-gases go through roughing filters and HEPA filters (95 percent efficient) before release to the atmosphere.

In the DC facility, process gas passes through sintered metal filters (nickel) with trapped solids returned to process; off-gases continue to KOH scrubbers (for HF removal), then are diluted (for H removal) and finally released to the atmosphere. Ventilation off-gases pass through roughing filters, HEPA filters (95 percent efficient) and are released.

2.1B.3 Emissions of Radionuclides

Some reported uranium effluents for specific fuel fabrication plants are given in table 2.1B-2. The data serve to indicate order-of-magnitude values, but many reports do not relate the effluent data to the quantities processed.

2.1B.4 Model Fuel Fabrication Facility

A model facility (table 2.1B-3) has been developed for assessing the impact of uranium fuel fabrication plants. Facility production values were based on 1500 MTU/y as reported by Pechin (PE75). The maximum exposed individual was assumed to be located 500 meters downwind.

Other operating parameters (such as control technology) were taken from the ORNL model (PE75) and were assumed to be the same for both ADU and DC type facilities.

Estimated radioactive emissions for the model facilities are shown in table 2.1B-4.

Table 2.1B-2. Atmospheric emissions of radionuclides from currently licensed uranium fuel fabrication facilities

Plant	Emissions (Ci/y)
Exxon (JNC70)	0.00015
Kerr-McGee (KMC71)	0.04
General Electric (Ly78, GEC71)	0.0028
Westinghouse (WEC72)	0.2
Nuclear Fuel Services (NRC78)[a]	0.01
Babcock & Wilcox (NRC78)[a] (Lynchburg, Virginia)	0.0007

[a] Effluent data reported for January 1 to June 30, 1977, were doubled to estimate an annual release rate.

Table 2.1B-3. Model uranium fuel fabrication facility

Parameter	Value
Type of facility:	
Ammonium diuranate (ADU)	UF_6 feed to plant hydrolyzed in water, uranium precipitated in ammonia to form ADU. ADU calcined to form UO_2.
Direct conversion (DC)	UF_6 feed to plant reacted with water vapor and hydrogen to form UO_2.
Annual capacity	1500 metric tons of uranium
Stack	10 meters fixed stack height with no plume rise

Table 2.1B-4. Atmospheric emissions of radionuclides from the model uranium fuel fabrication facility

Radionuclide	Emissions (Ci/y)	
	ADU	DC
Uranium-234	7.6E-3	5.9E-3
Uranium-235	2.6E-4	2.0E-4
Uranium-236	3.9E-4	3.0E-4
Uranium-238	9.7E-4	7.5E-4
Thorium-231	2.6E-4	2.0E-4
Thorium-234	9.7E-4	7.5E-4
Protactinium-234	9.7E-4	7.5E-4

ADU Ammonium diuranate. DC Direct conversion.

2.1B.5 Health Impact Assessment for Model Uranium Fuel Fabrication Facility

Annual radiation doses due to airborne radioactive emissions from model ADU and DC fuel fabrication facilities are presented in table 2.1B-5. These estimates are for a site located in the suburbs of a large Midwestern city (Site B, Appendix A). The maximum individual dose equivalent rate occurred 500 meters downwind.

Table 2.1B-6 estimates the individual lifetime fatal cancer risks and committed fatal cancers to the regional population. The lifetime fatal cancer risk to the highest exposed group of individuals is approximately 1.5E-5 for the ADU facility and 1.2E-5 for the DC facility. The lifetime fatal cancer risk to the average individual in the region is estimated to be 8.8E-9 for the ADU facility and 6.5E-9 for the DC-facility.

The estimated number of fatal cancers per year of site operation to the regional population is estimated to be 3.0E-4 for the ADU facility and 2.3E-4 for the DC facility.

Table 2.1B-5. Annual radiation doses from radioactive emissions from model ADU and DC process fuel fabrication facilities

Organ	ADU Process Maximum individual (mrem/y)	 Average individual (mrem/y)	 Population (person-rem/y)
Lung	4.9	2.9E-3	7.0
Bone	4.6E-1	1.1E-4	2.7E-1
Kidney	1.2E-1	3.5E-5	8.6E-2
Liver	5.6E-2	2.5E-5	6.3E-2
Thyroid	7.8E-2	4.0E-5	1.0E-1
G.I. Tract	5.9E-2	2.1E-5	5.1E-2
Other soft tissue	7.1E-2	3.6E-5	8.9E-2

Organ	DC Process Maximum individual (mrem/y)	 Average individual (mrem/y)	 Population (person-rem/y)
Lung	3.9	2.2E-3	5.5
Bone	3.6E-1	8.4E-5	2.1E-1
Kidney	9.2E-2	2.7E-5	6.7E-2
Liver	4.3E-2	2.0E-5	4.9E-2
Thyroid	6.1E-2	3.1E-5	7.8E-2
G.I. Tract	4.6E-2	1.6E-5	4.0E-2
Other soft tissue	5.6E-2	2.8E-5	7.0E-2

ADU Ammonium diuranate. DC Direct conversion

Table 2.1B-6. Individual lifetime risks and population fatal cancers due to radioactive emissions from model ADU and DC process uranium fuel fabrication plants

Source	Individual lifetime risks Maximum individual	 Average individual	Expected fatal cancers per year of operation (Fatal cancers)
ADU process	1.5E-5	8.8E-9	3.0E-4
DC process	1.2E-5	6.5E-9	2.3E-4

REFERENCES

AEC74 U.S. Atomic Energy Commission, 1974, Environmental Survey of the Uranium Fuel Cycle, WASH-1248 (Directorate of Licensing, Fuels and Materials, U.S. Atomic Energy Commission, Washington, DC).

EPA73 Office of Radiation Programs, 1973, Environmental Analysis of the Uranium Fuel Cycle - Part I - Fuel Supply, EPA-520/9-73-003-C (U.S. Environmental Protection Agency, Office of Radiation Programs, Washington, DC).

EPA77 U.S. Environmental Protection Agency, 1977, Subchapter F-Radiation Protection Programs, Part 190 - Environmental Radiation Protection Standards for Nuclear Power Operations, Federal Register, Vol. 42, No. 9, Thursday, January 13, 1977.

GEC71 General Electric Company, 1971, letter dated November 29, 1971, USAEC Docket No. 70-1113.

JNC70 Jersey Nuclear Company, 1970, Applicant's Environmental Report, Uranium Oxide Fuel Plant, No. JN-14, USAEC Docket No. 70-1257.

KMc71 Kerr-McGee Corporation, 1971, letter dated October 11, 1971, USAEC Docket No. 70-1113.

Ly78 Lyon R. J., Shearin R. L., and Broadway J. A., 1978, A Radiological Environs Study at a Fuel Fabrication Facility, EPA-520/5-77-004 (U.S. Environmental Protection Agency, Office of Radiation Programs, Washington, DC).

NRC78 U.S. Nuclear Regulatory Commission, 1978, letter from H. T. Peterson, NRC, to T. W. Fowler, EPA with enclosure of special nuclear material license effluent reports.

NRC79 U.S. Nuclear Regulatory Commission, 1979, Program Summary Report, NUREG 0380, Vol. 3, No. 3, March 16, 1979.

2.1C Light-Water Reactor Nuclear Power Plants

2.1C.1 General Description

As of June 30, 1978, there were 70 operable nuclear power reactors in the United States; 91 were being built and 42 were planned (DOE78). With only two exceptions (an operable high temperature gas-cooled reactor and a planned breeder reactor), all of the nuclear power reactors are either boiling water reactors (BWR) or pressurized water reactors (PWR). Pressurized water reactors comprise approximately two-thirds of the light-water generating capacity and it is assumed this two-to-one PWR-BWR ratio will continue through the year 2000.

Projections of electrical generation by nuclear energy in the United States are difficult to estimate for a number of reasons--uncertainty regarding demands for electricity, the availability of alternative fuels, and national policy regarding nuclear power. However, projections for the United States by the Department of Energy estimated an installed nuclear capacity in the year 2000 to range from 255 GW(e) (low case) to 395 GW(e) (high case)(C178).

2.1C.2 Process Description

A light-water-cooled nuclear power station operates on the same principle as a conventional fossil-fueled (oil or coal) power station except that the heat generation is provided by nuclear fission rather than combustion. The heat liberated in either process is used to convert water into steam. The steam enters a turbine which is connected to a generator that produces alternating electric current. The fission process in the nuclear power plant produces radioactive gases in the fuel (fission products) and also in the coolant through the absorption of neutrons by coolant and structural materials (activation products). A typical power reactor is expected to experience a certain amount of fuel failures or defects over its operating life through which radioactive gases may escape the fuel rod and enter the coolant. As a result, radioactive fission gases are present to some extent in the coolant of the reactor at all times.

The major gaseous radionuclide emissions for the BWR result from noncondensable off-gas removed from the main turbine condenser by the steam jet air ejector (SJAE). Gaseous waste from the SJAE enters a holdup system, to allow for decay of short-lived radionuclides, and is then discharged to the atmosphere. Even after holdup, the activity discharged from the SJAE is typically the predominant source of gaseous radioactive waste at a BWR.

The major source of gaseous radioactivity at a PWR results from degassing of the coolant and shim bleed operations which are designed to remove dissolved radioactive and stable gases in the coolant. These gases then enter a gaseous waste system and are transferred to a holdup system for decay of short-lived radionuclides before being released to the environment. At a PWR, gaseous wastes from other sources (e.g., building ventilation) may contribute comparable amounts of radioactivity.

2.1C.3 Emissions of Radionuclides

EPA evaluated LWR source terms, control technology and projected health impacts for different levels of effluent treatment systems (EPA73, EPA76) in developing the uranium fuel cycle standard (40 CFR 190). EPA has also summarized effluent data from light-water-cooled nuclear power plants from 1973-1976 (Ph77). Tables 2.1C-1 and 2.1C-2 summarize the airborne releases from single-unit BWRs and PWRs for 1976. Variation in annual releases may be attributed to plant operating parameters such as operating time, power levels, control technology, and equipment malfunction.

Table 2.1C-1. Radioactive emissions from operating single-unit BWRs, 1976

Facility	Emissions (Ci/y) Noble gases	Tritium	Halogens
Dresden 1	460,000	61.3	2.34
Big Rock Point	15,200	7.8	.02
Humboldt Bay 3	93,000	1.3	.37
La Crosse	124,000	12.6	.10
Oyster Creek 1	166,000	1.1	46.4
Nine Mile Point 1	176,000	18.5	8.6
Millstone Point 1	507,000	28.7	36.5
Monticello	11,400	76.9	1.0
Vermont Yankee	2,870	14.2	.1
Pilgrim 1	183,000	37.4	2.0
Cooper	38,100	67.1	.1
Duane Arnold	5,260	16.4	.1
Hatch 1	3,110	1.4	1.0
Fitzpatrick 2	46,200	15.4	5.8

Table 2.1C-2. Radioactive emissions from operating single-unit PWRs, 1976

Facility	Emissions (Ci/y)		
	Noble gases	Tritium	Halogens
Yankee (Rowe)	27.1	2.02	1.3E-4
San Onofre 1	417	47.2	4.5E-3
Haddam Neck	492	739	7.3E-4
Ginna, R.E.	5,520	23.6	3.4E-2
Robinson, H.B.	791	158	2.5E-1
Palisades	29.9	NR	3.1E-2
Maine Yankee	1,300	3.7	1.6E-3
Fort Calhoun	2,150	2.5	7.2E-1
Kewaunee	1,600	0.7	3.3E-3
Three Mile Island 1	2,760	717	8.6E-3
Arkansas One 1	5,690	6.7	4.1E-2
Rancho Seco	127	9.1	1.1E-3
Cook, Donald C.	975	.11	1.4E-3
Millstone Point 2	1,550	15.	4.9E-2
Trojan	527	1.5	3.7E-4
St. Lucie	1,790	2.1	2.0E-3
Beaver Valley 1	1.08	3,720	4.3E-6
Salem 1	8.9E-5	1.5E-5	ND

ND Not detectable. NR Not reported.

2.1C.4 Model Facility

Model facilities were developed for the impact analysis of BWRs and PWRs (table 2.1C-3). The model BWR and PWR characteristics (with recirculating u-tube type steam generators) as developed by the NRC are described in the final generic environmental statement on the use of recycled plutonium in mixed-oxide fuel for light-water cooled reactors (NRC76).

Emissions

Atmospheric emissions of radioactive materials from the NRC model facilities are presented in table 2.1C-4. These models are meant to illustrate typical cases and source terms and are not directly applicable to a particular operating reactor (NRC76).

2.1C.5 Health Impact Assessment of Model BWR and PWR Facilities

Radiological impacts of airborne effluents from light-water-reactors have been estimated using the AIRDOS-II computer code (Mo77) and emissions reported in table 2.1C-4. The maximum individual, average individual, and population dose equivalent rates are presented in table 2.1C-5. The estimates are for a site located in the suburbs of a large Midwestern city in the United States (Site B, Appendix A). Food production and consumption assumptions for the maximum individual were selected for a rural setting. The maximum individual dose equivalent rate occurred 503 meters downwind.

Table 2.1C-6 estimates the individual lifetime fatal cancer risks and committed fatal cancers to the regional population. The lifetime fatal cancer risk to the highest exposed group of individuals is approximately 2.0E-5 for the BWR and 8.6E-6 for the PWR. The lifetime fatal cancer risk to the average individual in the region is estimated to be 3.8E-8 for the BWR and 2.5E-8 for the PWR. The number of fatal cancers per year of site operation is estimated to be 1.3E-3 for the BWR and 8.8E-4 for the PWR to the regional population.

An assessment of the health impact from emissions of carbon-14 and tritium must also consider the impact to the population living beyond 80 kilometers since these radionuclides are dispersed worldwide. The number of fatal cancers committed to the world population from carbon-14 and tritium releases is estimated to be 3.7E-2 for the BWR and 2.2E-2 for the PWR (Fo79). These fatal cancers estimates are the impact over a 100-year period from one year of carbon-14 and tritium emissions. Risk factors used in estimating the fatal cancers committed to the world population from carbon-14 and tritium are presented in table 2.1C-7. The tritium risk coefficient was calculated using the UNSCEAR (UN77) estimated collective dose to the global population of 3×10^{-3} person rad per curie of tritium released to the atmosphere. A fatal cancer risk coefficient of 200 fatal cancers per 10^6 person-rem was utilized since tritium is uniformly distributed throughout body tissue. The carbon-14 risk coefficients employed estimates of the global impact of carbon-14 discharges to the atmosphere using the diffusion model of the carbon cycle developed by Killough (Ki77).

The 100-year global collective dose commitment factors for carbon-14 are 28 person-rem per curie to the total body and 10.7 person-rem per curie to the gonads. For 10^6 person-rem exposure to the total body from carbon-14, the following fatal cancers estimates were used: 58 leukemia deaths and 88 other cancer deaths. The health effects estimates are for carbon-14 beta irradiation only. No attempt was made to estimate the effect of carbon-14 to nitrogen-14 transmutation.

In addition, there is an impact to the world population from krypton-85; however, the impact is small compared to that of carbon-14. Ellett (E176) estimates 7.0E-5 committed fatal cancers for annual releases of krypton-85 from a 1000 MW(e) light water reactor.

Table 2.1C-3. Model BWR and PWR light-water-reactors

Parameter	Value
Type:	Boiling water reactor and Pressurized water reactor
Capacity:	1,000 MW(e)
Fuel:	Uranium only
Stack:	
Height	20-meter, fixed stack height with no plume rise

Table 2.1C-4. Atmospheric emissions of radionuclides from model BWR and PWR facilities (NRC76)

	Emissions (Ci/y)	
Radionuclide	BWR	PWR
Argon-41	25	25
Krypton-83m	(a)	1
Krypton-85m	150	16
Krypton-85	290	470
Krypton-87	200	3
Krypton-88	240	23
Xenon-131m	18	82
Xenon-133m	(a)	120
Xenon-133	3,200	12,000
Xenon-135m	740	(a)
Xenon-135	1,100	86
Xenon-138	1,400	(a)
Iodine-131	0.3	0.025
Iodine-133	1.1	0.023
Carbon-14	[b]9	[b]5
Tritium	43	1,100

[a] Ci/y=<1.
[b] Source: Fowler (Fo76).

Table 2.1C-5. Annual radiation doses from model BWR and PWR facilities

Organ	BWR Maximum individual (mrem/y)	Average individual (mrem/y)	Population (Person-rem/y)
Lung	1.5	2.9E-3	7.3
Bone	2.4	4.8E-3	12.0
Kidney	1.4	2.6E-3	6.5
Liver	1.5	2.7E-3	6.7
Thyroid	11.0	7.6E-3	19.0
G. I. tract	1.4	2.4E-3	6.0
Other soft tissue	1.8	3.3E-3	8.3

Organ	PWR Maximum individual (mrem/y)	Average individual (mrem/y)	Population (Person-rem/y)
Lung	6.0E-1	1.9E-3	4.6
Bone	1.2	3.7E-3	9.3
Kidney	5.9E-1	1.6E-3	4.1
Liver	6.0E-1	1.6E-3	4.1
Thyroid	1.4	2.7E-3	6.8
G. I. tract	5.7E-1	1.3E-3	3.3
Other soft tissue	7.5E-1	2.1E-3	5.3

Table 2.1C-6. Individual lifetime risks and population fatal cancers due to radioactive emissions from model BWR and PWR facilities

Source	Individual lifetime risks Maximum individual	 Average individual	Expected fatal cancers per year of operation (Fatal cancers)
BWR facilities	2.0E-5	3.8E-8	1.3E-3
PWR facilities	8.6E-6	2.5E-8	8.8E-4

Source	Expected total cancers in the worldwide population over the next 100 years[a] Tritium (Fatal cancers)	 Carbon-14 (Fatal cancers)	 Total (Fatal cancers)
BWR facilities	2.6E-5	3.7E-2	3.7E-2
PWR facilities	6.1E-4	2.1E-2	2.2E-2

[a] Calculated as described in Section 2.1C.5.

Table 2.1C-7. Tritium and carbon-14 risk coefficients for fatal cancers committed to the world population

Radionuclide	Fatal cancers committed to the world population per curie released to the atmosphere (Fatal cancers/Ci)
Tritium	6.0E-7
Carbon-14 (100 years)	4.1E-3

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EPA76 Office of Radiation Programs, 1976, 40 CFR 190 Environmental Radiation Protection Requirements for Normal Operations of Activities in the Uranium Fuel Cycle, Final Environmental Statement, Volume I and II, EPA 520/4-76-016 (U.S. Environmental Protection Agency, Office of Radiation Programs, Washington, DC).

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2.2 High Temperature Gas Cooled Reactor (HTGR)

2.2.1 General Description

A high temperature gas cooled power reactor uses uranium-235 as fissile fuel in the initial reactor core, thorium-232 as fertile material which is converted to uranium-233, graphite as the moderator, cladding structure and reflector, and helium as a coolant. The heat generated within the fuel element is transferred through the fuel cladding to the primary coolant, helium gas. The helium transfers heat to a secondary water system to produce steam. The resulting steam is used to turn a turbine generator to produce electricity.

The only operating commercial nuclear power plant using a gas cooled reactor system in the United States today is Fort St. Vrain near Greeley, Colorado, which started up in 1978. The first gas cooled reactor system in the United States was Peach Bottom 1 in Pennsyslvania which operated between 1966 and 1974 is now decommissioned. The proposed station in Delaware, Summit Power Units 1 and 2, has been cancelled.

2.2.2 Process Description

In operation, the gas cooled reactor system converts thorium-232 to uranium-233, a fissle material. The helium coolant flows through the reactor core to the steam generators, where it gives up heat to convert water to steam, and then returned to the reactor by helium circulators. The generated steam flows to the turbine to produce electricity. The reactor, with its accessories (helium circulators and steam generators), is contained in a prestressed concrete reactor vessel (PCRV), a vertical hexagonal prism, 19 by 32 meters, with three-foot thick side walls.

The principal source of high activity gaseous waste originates from the helium purification system. Small amounts of potentially contaminated gaseous waste also come from the sampling of the primary coolant, purging of fuel storage and handling systems, purging of the helium circulator handling-cask and from the PCRV support floor vent and liquid waste tank vent headers. Gas enters the gaseous waste system through either of two paths, the low or high activity vent headers. The low activity vent header collects gases of sufficiently low activity and flow rate that they can be discharged through the reactor building vent after passing through a prefilter, a high efficiency particulate filter (HEPA), and a charcoal adsorber.

Table 2.2-1. Estimated atmospheric emissions of radionuclides from a typical gas cooled reactor system

Radio-nuclide	Emissions[a] (Ci/y)				
	Reactor building leakage	Regeneration system	Air ejection	PCRV leak[b]	Total
Krypton-85	0.05	947	0.002	-	947
Krypton-87	2.0	-	1.2	25.6	29
Krypton-88	11.0	-	3.2	34.9	49
Xenon-131m	0.04	4.8	0.002	-	5
Xenon-133	12.0	9.0	0.6	.4	22
Xenon-135	2.4	-	2.4	13.	16
Xenon-138	0.01	-	0.03	7.1	7
Total					1075

[a] Regeneration System--Based on minimum holdup time of 60 days.
[b] PCRV Leak--Prestressed concrete reactor vessel. Numbers were extrapolated from AEC74 by multiplying values by 0.42.

The high activity header collects gases that are normally too radioactive to be released after treatment by filtration only. These gases are routed to the helium purification system which removes fission products and chemical impurities from the primary coolant. The system, which uses two alternately operating gas processing trains, consists of a high-temperature filter-adsorber to remove particulates and halogens (mostly iodines), a helium cooler, a dryer, low-temperature adsorber, and a hydrogen getter unit (titanium sponge) which removes gaseous hydrogen and tritium.

The helium dryer and the low-temperature filter-adsorber are regenerated by passing hot helium through the unit, which strips the accumulated gases (including the radioactive ones) from the adsorber. The gases are collected and analyzed prior to controlled venting to the reactor building vent. The anticipated annual releases of radioactive material in the regeneration off-gases shown in table 2.2-1 were based on a minimum holdup time of 60 days.

Before entering the reactor building vent, the effluent of both treatment systems passes through HEPA filters and charcoal adsorbers. If the activity is above a predetermined level, the gas stream is diverted to the vacuum tank of the high activity system for further treatment.

The stream in the economizer and superheater sections of the steam generator is at a higher pressure (about 2500 psia) than the helium (about 693 psia), so any leakage will be essentially steam into the primary coolant (helium). However, the steam in the reheat section of the steam generator is at a lower pressure (about 649 psia) so it is possible for radioactive primary coolant to enter the steam through leaks in this portion of the steam generator. Activity in the secondary system can be released via the air ejector.

2.2.3 Emissions of Radionuclides

Table 2.2-1 summarizes atmospheric emissions from the model plant for four release pathways as reported in final environmental impact statements for Fort St. Vrain (AEC72) and the proposed Summit Power Station (AEC74). The PCRV leakage source for Summit was scaled by power level to apply to the model plant. Emissions from the regeneration systems were based on a minimum holdup time of 60 days. There are no operating data available yet from Fort St. Vrain which began an operational testing period during 1978.

2.2.4 Typical Facility

Since there is only one gas cooled power reactor operating in the United States, the characteristics of that facility have been used to describe the typical plant in table 2.2-2 (AEC72).

2.2.5 Health Impact Assessment of Model Facility

The final environmental impact statement (AEC72) for Fort St. Vrain estimates an annual total body dose of 6.4 person-rems (table 2.2-3) from atmospheric emissions of radioactive materials to the population within a 50-mile radius of the site. Estimates are for a low population density site in Western United States. The population within 80 km is about 1.4 million. This dose, however, does not include the small contribution from PCRV leakage. The risk from this exposure is estimated to be .0013 fatal cancers per year. The majority of this exposure is from krypton-85 emissions from the regeneration system. Doses and risks are presented in table 2.2-3 and table 2.2-4.

Table 2.2-2. Typical high temperature gas cooled reactor system

Parameter	Value
Thermal power (MWth)	842
Electrical power (MWe)	332
Plant lifetime	30 years
Fuel cycle	U/Th
Fuel temperature (°F)	2300
Enrichment (% Uranium-235)	93
Fuel element material	Graphite
Coolant pressure (psia)	700
Coolant loops	2
Coolant cleanup system:	
Tritium	Titanium sponge
Noble gases	Low temperature adsorber
Containment building type	Confinement only
Radwaste system (gaseous)	
Type	Pressurized decay tank
Retention days	60

Table 2.2-3. Annual radiation doses due to atmospheric emissions from Fort St. Vrain (AEC72)

Source	Nearest resident (mrem/y)	Average individual (mrem/y)	Population (person-rem/y)
Fort St. Vrain	0.52	4.5E-3	6.4

Table 2.2-4. Individual lifetime risks and numbers of fatal cancers due to radioactive emissions from Fort St. Vrain

Source	Individual lifetime risks Nearby residents	 Average individual in region	Expected fatal cancers per year of operation (Fatal cancers)
Fort St. Vrain	5.5E-6	6.4E-8	1.3E-3

REFERENCES

AEC72 Atomic Energy Commission, 1972, Final Environmental Statement related to operation of the Fort St. Vrain Nuclear Generating Station of Public Service Company of Colorado, Docket No. 50-267, U.S. Atomic Energy Commission, August 1972.

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2.3 Radiopharmaceutical Industry

2.3.1 General Description

The radiopharmaceutical industry converts radioactive chemicals into a form suitable for use in medicine and research. The radiopharmaceutical industry is composed of producers and users of radiopharmaceuticals. Radioisotopes used in medical practice are produced directly in nuclear reactors, particle accelerators, and in isotope generators by parent-daughter separation. Producers include those companies which produce isotopes as raw materials for other pharmaceutical companies as well as the secondary companies which produce radiopharmaceuticals through tagging and labelling processes.

Radioactive materials are widely used for medical diagnosis and therapy. Since 1946, the number of medical institutions in the United States licensed to use radioactive materials derived from nuclear chain reactions has grown from 38 to more than 12,000, including both NRC and Agreement State licensees. These licensees perform an estimated 30 million nuclear medicine procedures per year at an estimated cost of $1.6 billion (NRC77).

Producers

Table 2.3-1 lists the principal producers of radiopharmaceuticals in the United States (AEC74). A more current list should be available in late 1979 when an EPA contract requesting this information is completed.

Approximately 12 commercial organizations are currently providing either cyclotron- or reactor-produced radioisotopes (AEC71). The Federal government is gradually relinquishing to industry its role as prime supplier of reactor-produced radionuclides.

Private industry now produces 65 different, generally used radioisotopes, as compared to about 50 low-volume, special-use items still produced in Federal installations. For practical considerations, therefore, the Federal government is not being considered as a significant supplier of medical radioisotopes. The principal industrial suppliers of cyclotron- and reactor-produced radioisotopes are listed in table 2.3-2.

Table 2.3-1. Principal industrial processors of organic labeled compounds, radiochemicals and radiopharmaceuticals (AEC74)

Processor	Radio-chemicals	Radio-immuno-assay reagents[a]	Radio-pharma-ceuticals
Abbott Laboratories North Chicago, IL	-	X	X
Aerotest Operations, San Ramon, CA	X	-	-
American Radiochemical Corp Sanford, FL	X	-	-
Amersham/Searle, Arlington Heights, IL	X	X	X
Ames Company, Division of Miles Laboratories, Inc., Elkhart, IN	X	-	X
Bio-Chemical & Nuclear Corp., Burbank, CA	X	-	-
Bio-Rad Laboratories Richmond, CA	X	-	-
Calatomic, Inc. Los Alamos, CA	X	-	-
California Radiochemicals, Inc., Los Angeles, CA	X	-	-
Cambridge Nuclear Corp., Billerica, MA, and Princeton, NJ (Subsidiary of NL Industries, Inc.)	X	-	X
Curtis Nuclear Corp., Los Angeles, CA	X	-	X
Dhom Products, Ltd., North Hollywood, CA	X	-	X

[a]See footnote at end of table.

Table 2.3-1. Principal industrial processors of organic labeled compounds, radiochemicals and radiopharmaceuticals--continued

Processor	Radio-chemicals	Radio-immuno-assay reagents[a]	Radio-pharma-ceuticals
Diagnostic Isotopes, Inc. Upper Saddle River, NJ	-	-	X
Virgo Reagents, Electro-Nucleonics Lab, Bethesda, MD	-	X	-
General Electric Company, Pleasanton, CA	X	-	-
Gamma Industries, (Div. of Nuclear Systems, Inc.) Houston, TX, and Baton Rouge, LA	X	-	-
Imaj International, Inc., (Nuclear Medicine Div of Allergan Pharmaceuticals) Irvine, CA	-	-	X
Industrial Nuclear Company, Inc., Overland, MO	-	-	X
International Chemical & Nuclear Corp., Irvine, CA	X	-	-
Isolab, Inc., Akron, OH	X	-	-
Iso-Med, Inc., Hawthorne, CA, (Div of New England Nuclear)	-	-	X
Mallinckrodt Chemical Works, St. Louis, MO	X	-	X
Medi-Physics, Emeryville, CA, and South Plainfield, NJ	-	-	X
Miles Laboratories, Inc., Elkhart, IN	X	-	-
New England Nuclear North Billerica, MA	X	-	X

[a] See footnote at end of table.

Table 2.3-1. Principal industrial processors of organic labeled compounds, radiochemicals and radiopharmaceuticals--continued

Processor	Radio-chemicals	Radio-immuno-assay reagents[a]	Radio-pharma-ceuticals
Nuclear Associates, Inc. Westburg, NY.	X	-	-
Nuclear Dynamics, Elmonte, CA	X	-	-
Nuclear Equipment Chemical Corp., Farmingdale, NY	X	-	-
Nuclear Medical Labs, Inc., Dallas, TX	X	-	-
Schartz/Mann, Div. of Becton, Dickinson & Co., Orangeburg, NY	X	X	-
E. R. Squibb & Sons, New Brunswick, NJ	-	X	X
Teledyne Isotopes, Palo Alto, CA	X	-	-
Union Carbide Corp., Tuxedo, NY	X	-	-
Worthington Biochemical Corp., Freehold, NJ	-	X	-

[a] Radioimmunoassay reagents are organic labeled compounds which are used in isotope dilution techniques for the sensitive measurement of biological components--for example, the measurement of hormone levels in blood plasma.

Users

Radiopharmaceutical users are defined as hospitals or private physicians using radionuclides for medical purposes including the sewage treatment plants which receive and treat the liquid wastes.

Diagnostic and therapeutic procedures using radionuclides are listed in table 2.3-3. Data from a Public Health Service survey indicate that one in every four patients admitted to hospitals is given a radioactive tracer as part of his medical diagnostic workup (BRH70). The rapid growth of nuclear medicine is partly due to the increasing availability of radiopharmaceuticals in more easily administered forms.

Diagnostic nuclear medicine includes such techniques as measuring the uptake of radioactive drugs by individual organs (for such purposes as assessing thyroid function), "imaging" the distribution of radioactive drugs among organs or within an organ (to detect the presence of tumors, for example), estimating the size of certain body pools (such as red blood cell and blood plasma volumes), and measuring the components in biological samples (such as protein binding sites and hormones in blood and urine) (NRC77).

Therapeutic techniques include the use of radioactive drugs internally (for example, in the treatment of thyroid cancers), the use of radioactive devices both as implants and on the surface of the body (termed "brachytherapy," or "therapy from a short distance") and the use of radioactive devices external to the body (termed "teletherapy," or "therapy from a distance")(NRC77).

Radionuclides are also used in research studies to obtain basic medical data and to develop a clinical diagnostic or therapeutic procedure. There is no current information available on research studies using medical isotopes. A 1966 BRH survey reported the most frequently conducted research studies at that time. These studies (table 2.3-4) accounted for 57 percent of all research studies reported (BRH70).

2.3.2 Process Description

Radioisotopes used by the radiopharmaceutical industries are produced by accelerators, by small research reactors, and by radioisotope generators.

Table 2.3-2. Principal industrial suppliers of cyclotron- and reactor-produced radioisotopes

Principal Industrial Suppliers of Cyclotron Radioisotopes

Amersham-Searle, Arlington Heights, Illinois
Cambridge Nuclear Corporation, Billerica, Massachusetts
Diagnostic Isotopes, Upper Saddle River, New Jersey
International Chemical and Nuclear Corporation, Irvine, California

Mallinckrodt Chemical Works, St. Louis, Missouri
Medi-Physics Inc., Emeryville, California
New England Nuclear Corporation, Billerica, Massachusetts

Commercially Available Sources of Reactor-Produced Radioisotopes

The following facilities offer irradiation services for isotope production or other irradiation testing:

Northrup Corporate Labs
TRIGA Mark-F
Hawthorne, California

University of Michigan
Ann Arbor, Michigan

Western New York Nuclear
Research Center, Inc.
Pulstar
Buffalo, New York

Union Carbide Corp., UCNR
Sterling Forest, New York

University of Missouri
Columbia, Missouri

Table 2.3-3. Major radiopharmaceuticals and their uses (FDA76, NRC79)

Radionuclide	Use
Phosphorus-32	Bone marrow therapy
Gallium-67	Tumor localization
Rubidium-81	Myocardial imaging
Technetium-99m	Bone imaging, brain imaging, liver imaging, lung perfusion, myocardial imaging, blood pool, renograms, thyroid imaging, thyroid uptake renal imaging
Iodine-123	Thyroid imaging Thyroid uptake
Iodine-125	Renograms
Iodine-131	Renal imaging, renograms, thyroid imaging, thyroid uptake, tumor localization and therapy
Xenon-133	Lung ventilation
Mercury-203	Renograms
Thalium-201	Myocardial imaging

Table 2.3-4. Radioisotopes used in research (BRH70)

Radionuclide	Compound	Study
Iodine-131	Labeled hormones	Metabolic
Tritium and carbon-14	Labeled aldosterone	Metabolic
Iodine-131	Labeled albumin	Blood volume
Carbon-14	Labeled steroids	Metabolic
Tritium	Labeled cholesterol	Metabolic
Tritium	Labeled aldosterone	Adrenal secretion
Xenon-133	Saline solution	Blood flow
Technetium-99m	Labeled albumin	Lung scanning
Krypton-85	Gas	Brain blood flow
Xenon-133	Gas	Muscle blood flow
Krypton-85	Saline solution	Cardiac output
Tritium	Labeled thymidine	Autoradiography
Iron-59	Ferric chloride	Iron absorption
Iodine-131	Labeled albumin (MAA)	Lung scanning
Carbon-14	Labeled cortisol	Plasma clearance
Calcium-47	Calcium chloride	Metabolic
Indium-113m	Iron complex	Lung scanning
Tritium	Labeled steroids	Metabolic
Iodine-131	Labeled human F.S.H.	Radioimmunoassay
Technetium-99m	Labeled albumin	Heart scanning

2.3.2 Process Description

Radioisotopes used by the radiopharmaceutical industries are produced by accelerators, by small research reactors, and by radioisotope generators.

Reactor Produced

Most radioisotopes are made in nuclear reactors by one of the reactions shown in table 2.3-5. The most common is the neutron-gamma reaction because many elements capture neutrons easily. Such radioisotopes as ^{24}Na, ^{59}Fe, ^{60}Co, and ^{198}Au (McG77) are produced by neutron capture.

Table 2.3-5. Nuclear reactions used in radioisotope production

Reaction	Examples
(1) Neutron-gamma (n,γ)	$^{59}Co + n \rightarrow {}^{60}Co + \gamma$
(2) Neutron-proton (n,p)	$^{32}S + n \rightarrow {}^{32}P + p$
(3) Neutron-alpha (n,α)	$^{35}Cl + n \rightarrow {}^{32}P + \alpha$

The main steps (Ba66) in the production of radionuclides in a reactor are:

1. A suitable target is prepared and irradiated with neutrons.
2. The irradiated target is processed by simple dissolution or by more complicated separations--including ion exchange, precipitation and distillation--to remove undesirable impurities or to concentrate the product nuclide.
3. The radionuclides are placed in inventory, dispensed, and packaged for shipment.

Accelerator Produced

A wide variety of radioisotopes are produced in particle accelerators, such as the cyclotron. The amount of radioactive material which can be produced in particle-accelerating machines is smaller than that made in a nuclear reactor.

The cyclotron is used to produce two principal types of nuclides. First, those whose decay characteristics are preferable to other radioisotopes of the same element that are available from nuclear reactors; and second, isotopes of elements of biological importance for which no reactor-produced nuclides exist. Examples of the first category are: iodine-123, iron-52, mercury-199m, chromium-49, copper-61 and selenium-73; examples of the second are carbon-11, nitrogen-13, and oxygen-15.

Generator Produced

Because of the difficulties encountered in the preparation and delivery of radionuclides with short half-lives, there is a growing trend for hospitals to operate radioisotope generators for the production of certain isotopes, notably technetium-99m.

These devices make short-lived nuclides available at long distances from the source of production. They consist of a longer-lived parent nuclide that produces the short-lived daughter nuclide as it decays. The daughter nuclide is chemically separated at intervals and the parent is left to generate a fresh supply of the daughter (Wa68).

Nuclides that have the potential to be produced by radioisotope generators are listed in table 2.3-6.

Table 2.3-6. Potential generator systems from AEC Symposium No. 6 (AEC66)

Daughter isotope	Half-life	Parent isotope	Half-life
Arsenic-72	26 hours	Strontium-72	8.4 days
Cesium-131	9.7 days	Barium-131	11.6 days
Indium-113m	1.7 hours	Tin-113	118 days
Potassium-42	12.4 hours	Argon-42	>3.5 days
Praseodymium-144	17.3 minutes	Cerium-144	285 days
Rhodium-103m	57 minutes	Palladium-103	17 days
Scandium-44	3.9 hours	Titanium-44	$\sim 10^3$ years
Technetium-99m	6.0 hours	Molybdenum-99	66.0 hours
Tellurium-125m	58 days	Strontium-125	2.7 years
Tellurium-127	9.3 hours	Tellurium-127m	105 days

2.3.3 Emissions of Radionuclides

Radiopharmaceutical Production Plants

Three medical isotope producers have submitted current emission data to EPA. All three firms are large in terms of production and volume sales, and the three firms together are representative of the major categories of medical isotope producers, i.e., radiochemical, radioimmunoassay reagent, or radiopharmaceutical producer. The emission data represent airborne releases from normal operations of the facility as measured by company-owned monitoring systems. The average annual releases are summarized in table 2.3-7.

Table 2.3-7. Atmospheric emissions of radionuclides from three major radiopharmaceutical producers

Source	Emissions (Ci/y)		
	Plant X	Plant Y	Plant Z
Iodine-125	9.4E-1	2.6E-3	2.7E-2
Iodine-131	8.8E-1	3.1E-2	5.7E-1

Hospitals

A study conducted under EPA Contract 68-01-5049 (Te79) showed that approximately 3,000 hospitals are licensed by the NRC to administer radioisotopes for diagnostic and/or therapeutic purposes (Te79). This study indicates that iodine-131 and xenon-133 are released as airborne emissions due to normal preparation and administration procedures. Technetium-99m is not released. Source terms were estimated to be 1E-5 Ci/y for iodine-131 and 2 Ci/6 for xenon-133.

Sewage Treatment Plants

Most radioisotope releases from hospitals occur via the liquid pathway. When sewage containing radioisotopes is then treated in facilities employing sludge drying and combustion, radionuclides can be emitted into the air. EPA contract 68-01-5049 (Te79) studied the air emissions from a sewage treatment plant and estimated airborne source terms of 5E-4 Ci/y for iodine-131, and 8E-4 Ci/y for technetium-99m. The study further estimated that approximately 21 percent of sewage treatment facilities (about 4000) in the United States use incineration/pyrolysis for sludge treatment.

2.3.4 Typical Facility

Radiopharmaceutical Production Plant

Demographic, meteorological and emission parameters were developed from actual site characteristics of one of the plants in table 2.3-7 and by using rural food source parameters for Sites C-F in table A-2, Appendix A. All of the major medical isotope production facilities are located in urban settings with fairly high population densities.

The typical facility produces technetium-99m, xenon-133, iodine-131, iodine-125, and molybdenum-99/technetium-99m generators using a reactor-produced neutron activation process and discharges all airborne releases from a single stack. Atmospheric emissions from the typical facility are listed in table 2.3-9.

Production figures are considered proprietary but preliminary figures (Te79) estimate that the entire industry produces 800 to 1350 curies per year of iodine-131, 1600 to 3000 curies per year of xenon-133, and 15,600 to 30,600 curies per year of technetium-99m. It is reasonable to assume, therefore, that the typical facility (because it is one of the three largest) produces one-third of the entire industry's output. We have assumed production values of 350 curies per year of iodine-131, 800 curies per year of xenon-133, and 23,000 curies per year of technetium-99m.

The typical facility employes charcoal bed and HEPA filters as gaseous effluent control technology; annual emissions after such treatment are 0.94 curies of iodine-125 and 0.88 curies of iodine-131.

Hospital

The typical hospital facility is a general care hospital of more than 500 beds located in the suburbs of a large Midwestern city

(Site B, Appendix A). It is assumed that the hospital has nuclear medicine capabilities and administers an average of 0.6 curies per year of iodine-131 and 2.7 curies per year of xenon-133.

Radiopharmaceuticals are prepared in exhaust hoods in the nuclear medicine section; the exhaust from the building ventilation system enters the atmosphere without further treatment.

Diagnostic administrations occur in patients' rooms while therapeutic administrations occur in the nuclear medicine department. Emissions may be collected by the hospital ventilation system and enter the atmosphere without further treatment. Atmospheric emissions are listed in table 2.3-9.

Sewage Treatment Facility

The typical sewage treatment plant developed for assessing health impacts is located in the suburbs of a large Midwestern city (Site B, with rural food source fractions, Appendix A). The plant incinerates the sludge after it has been dried. Atmospheric emissions are listed in table 2.3-9.

Table 2.3-8. Model facilities of typical producers and users of radiopharmaceuticals

Parameter	Value
Radiopharmaceutical production plant	
Product line:	Iodine-131, xenon-133, technetium-99m, molybdenum-99/technetium-99m generators
Product volume:	Iodine-131 - 350 Ci/y Xenon-133 - 800 Ci/y Technetium-99m - 23,000 Ci/y
Emission control:	Charcoal/HEPA filters with release through a single elevated stack
Stack:	
Height	15.2 meters
Diameter	1.22 meters
Effluent velocity	20.1 meters/s
Rate of heat emission	0.0 cal/s
Hospital	
Size:	500+ beds
Volume of administrations:	Iodine-131 - 0.6 Ci/y Xenon-133 - 2.7 Ci/y
Emission control:	Exhaust hoods in nuclear medicine department; release through building ventilation system. In patients' rooms, release is through building ventilation system.
Sewage treatment plant	
Process:	Incineration following sludge drying.

Table 2.3-9. Atmospheric emissions of radionuclides from typical facilities of producers and users of radiopharmaceuticals

Radionuclide	Emissions (Ci/y)
Radiopharmaceutical production plant	
Iodine-125	9.4E-1
Iodine-131	8.8E-1
Hospital	
Iodine-131	1.0E-5
Xenon-133	2.0
Sewage treatment plant	
Iodine-131	5.0E-4
Technetium-99m	8.0E-4

2.3.5 Health Impact Assessment of Producers and Users of Radiopharmaceuticals

Table 2.3-10 presents estimates of annual radiation doses resulting from radioactive emissions from a typical radiopharmaceutical production plant, a typical hospital administering radioisotopes for diagnostic and/or therapeutic purposes, and a typical sewage treatment plant receiving and treating liquid wastes.

Estimated individual lifetime risks and the number of fatal cancers to the population resulting from doses from these facilities are shown in table 2.3-11.

Because the highest doses from these facilities is to the thyroid, the fatal cancer risk is primarily a risk of fatal thyroid cancer. The risk of nonfatal cancer of the thyroid is approximately an order of magnitude higher.

Table 2.3-10. Annual radiation doses due to radioactive emissions from typical facilities of producers and users of radiopharmaceuticals

Organ	Maximum Individual (mrem/y)	Average individual (mrem/y)	Population (peron-rem/y)
Typical radiopharmaceutical production plant			
Lung	1.5E-1	3.1E-5	4.7E-1
Bone	2.9E-1	6.2E-5	9.3E-1
Kidney	1.5E-1	3.2E-5	4.8E-1
Liver	1.3E-1	2.7E-5	4.0E-1
Thyroid	8.6E+1	1.3E-2	2.0E+2
G.I.tract	9.0E-2	1.9E-5	2.8E-1
Other soft tissue	2.0E-1	4.3E-5	6.5E-1
Typical hospital			
Lung	7.1E-5	2.9E-7	7.1E-1
Bone	1.5E-4	6.1E-7	1.5
Kidney	6.1E-5	2.4E-7	6E-1
Liver	5.9E-5	2.4E-7	5.9E-1
Thyroid	1.3E-4	5.3E-7	3.8E-1
G.I. tract	3.9E-5	1.5E-7	1.3
Other soft tissue	7.7E-5	3.1E-7	7.7E-1
Typical sewage treatment plant			
Lung	4.5E-5	2.6E-8	6.3E-2
Bone	5.6E-5	3.3E-8	8.1E-2
Kidney	4.2E-5	2.4E-8	6E-2
Liver	4.1E-5	2.3E-8	5.6E-2
Thyroid	2.6E-2	5.8E-6	1.4E+1
G.I. tract	3.7E-5	2.0E-8	4.9E-2
Other soft tissue	4.7E-5	2.7E-8	6.6E-2

Table 2.3-11. Individual lifetime risks and number of fatal cancers due to radioactive emissions from typical facilities of producers and users of radiopharmaceuticals

Source	Individual lifetime risks[a] Maximum individual	Average individual	Expected fatal cancers per year of operation[a] (Fatal Cancers)
Typical production plant	8.1E-06	4.4E-10	2.9E-4
Typical hospital	9.4E-10	3.7E-12	1.3E-7
Typical sewage treatment plant	2.3E-09	7.0E-13	2.5E-8

[a] The fatal cancer risk is primarily a risk of fatal thyroid cancer. The risk of nonfatal thyroid cancer is approximately and order of magnitude higher.

REFERENCES

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NRC77 Nuclear Regulatory Commission, 1977, Annual Report--1977, NRC, Washington, D.C.

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2.4 Test Reactors

2.4.1 General Description

The Department of Energy divides this category into three types: 1) general irradiation test reactors, 2) high-power research and test reactors, and 3) safety research and test reactors.

A general irradiation test reactor is defined as having: 1) a thermal power level exceeding 10,000 kW, 2) test loops or experimental facilities within or in proximity to the core, and 3) the use of nuclear radiation for testing the life or performance of reactor components as its major function.

A high-power research and test reactor is defined as having relatively high thermal power level (5000 kW or more) but not classified as a general irradiation test reactor.

A safety research and test reactor is defined as a reactor associated with nuclear safety research or engineering-scale test program conducted for the purpose of developing basic information or demonstrating safety characteristics of terrestrial and aerospace nuclear reactor systems (DOE78).

As of June 30, 1978, there were 15 reactors of this category in operation within the United States (DOE78). Two of these reactors, the National Bureau of Standards reactor and the Union Carbide Corporation reactor, are facilities not operated by the Department of Defense or Department of Energy (table 2.4-1).

The National Bureau of Standards reactor (NBSR) complex is located northwest of Washington, D.C., near Gaithersburg, Md. The reactor is an enriched-uranium, heavy-water cooled and moderated vessel type unit. The facility began operation in 1967 and has an available power capacity of 10 M watts. The high fluxes generated by the NBSR are used primarily to measure fundamental properties of matter and to develop new standards and measuring techniques.

2.4.2 Process Description

Test reactors have been constructed and operated in the United States since the early 1950's. Several types of test reactors, such as heavy water, graphite, tank and pool, have been constructed and used. These reactors have been used primarily to test new reactor designs, reactor components, safety features and to develop material standards.

Table 2.4-1. Test reactors (other than DOD or DOE)

Facility	Type	Power	Startup date
National Bureau of Standards Gaithersburg, Maryland	Heavy water	10 MW	1967
Union Carbide Corp.[a] Tuxedo, N. Y.			

[a] This reactor is used to produce radiopharmaceuticals and is discussed in section 2.3 of this report.

Table 2.4-2. Atmospheric emissions from test reactors (other than DOD or DOE facilities)

Facility	Emissions (Ci/y)
National Bureau of Standards	
Tritium	155
Argon-41	465
Union Carbide	(a)

NR Not reported.

[a] This reactor is used to produce radiopharmaceuticals and is discussed in section 2.3 of this report.

2.4.3 Emissions of Radionuclides

The airborne emission of radioactive materials from test reactors is usually confined to argon-41, tritium, and small quantities of fission products and noble gases.

The National Bureau of Standards reactor facility released approximately 155 curies of tritium and 465 curies of argon-41 to the environment as gaseous waste in 1977 (table 2.4-2). These were the only detectable radioisotopes released (To78).

2.4.4 Typical Facility

To analyze the impact of test reactors, the parameters of a typical facility (table 2.4-3) have been used in projecting population dose rates. The atmospheric emissions of radioactive materials are listed in table 2.4-4.

2.4.5 Health Impact Assessment of Typical Facility

Table 2.4-5 presents estimated annual radiation doses resulting from radioactive emissions from the typical test reactor facility. These estimates are for a site located in the suburbs of a large Midwestern city (Site B, Appendix A).

Individual fatal cancer risks and committed population fatal cancers are presented in table 2.4-6. The lifetime fatal cancer risk to the maximum exposed group of individuals is estimated to be about 2.4E-6. The lifetime fatal cancer risk to the average individual living within 80 kilometers of the typical facility is estimated to be 5.0E-9.

The estimated number of fatal cancers to the population living in the region around the facility is estimated to be 1.8E-4 per year of reactor operation. The world-wide fatal cancer commitment (tritium) is estimated to be 9.0E-5.

Table 2.4-3. Typical test reactor facility

Parameter	Value
Type of facility:	Enriched uranium, heavy water cooled and moderated test reactor
Power level:	10 M watt
Stack:	
Height	23.2 meters
Diameter	1.5 meters
Effluent velocity	12.7 meters/s
Rate of heat emission	0.0

Table 2.4-4. Atmospheric emissions of radionuclides from the typical test reactor facility

Radionuclide	Emissions (Ci/y)
Argon-41	450
Tritium	150

Table 2.4-5. Annual radiation doses due to radioactive emissions from a typical test reactor facility

Organ	Maximum Individual (mrem/y)	Average Individual (mrem/y)	Population (Person-rem/y)
Lung	2.1E-1	4.4E-4	1.1
Bone	2.1E-1	4.4E-4	1.1
Kidney	2.1E-1	4.4E-4	1.1
Liver	2.1E-1	4.4E-4	1.1
G.I. Tract	2.1E-1	4.4E-4	1.1
Thyroid	2.1E-1	4.4E-4	1.1
Other soft tissue	2.1E-1	4.4E-4	1.1

Table 2.4-6. Individual lifetime risks and number of fatal cancers due to radioactive emissions from a typical test reactor facility

Source	Individual lifetime risks		Expected fatal cancers per year of operation (Fatal cancers)
	Maximum individual	Average individual	
Typical facility	2.4E-6	5.0E-9	1.8E-4

Source	Expected total cancers in the worldwide population[a] over the next 100 years (Fatal cancers)
Typical facility	9.0E-5

[a] From 150 Ci/y of tritium.

REFERENCES

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2.5 Research Reactors

2.5.1 General Description

A research reactor, excluding those located at a university, uses nuclear radiation as a tool for basic or applied research, and has a thermal power level less than 5,000 kW (DOE78). As of June 1978, there were 24 research reactors operating in the United States. The 13 reactors not located in Department of Energy or Department of Defense facilities are listed in table 2.5-1 (DOE78).

2.5.2 Process Description

Several reactor designs are used in research reactors, as noted in table 2.5-1. The rated power output of these reactors ranges from negligible to 2,000 kW(t). A research reactor is generally designed to provide a source of neutrons and/or gamma radiation for research into basic or applied physics, biology, or chemistry or to aid in the investigation of the effects of radiation on materials.

2.5.3 Emissions of Radionuclides

The NRC regulations, under which all research reactors operate, do not require the submission of annual effluent release data. Therefore, there are no effluent release data for research reactors (NRC77). However, some licensees do submit release data to the NRC in conjunction with annual operating reports. All available data from NRC is shown in table 2.5-1 for 13 facilities (NRC77).

The variation in the emission data shown in table 2.5-1 may be attributed to parameters such as power level, operating time, and emission control systems which influence annual emissions for research reactors.

Because this category of reactors is used for a variety of research purposes, operating times and power levels vary accordingly. These parameters have not been compiled for each facility listed in table 2.5-1.

2.5.4 Model Facility

In order to estimate population and individual radiation doses, a model facility (table 2.5-2) was developed by assigning the various parameters that are important in assessing impacts.

Table 2.5-1. Research reactors (other than DOD or DOE)

Facility	Type	Power kW(t)	Startup date	Airborne Releases Argon-41 Ci	Airborne Releases Time Period
Aerotest Operations, San Ramon, Calif.	Pool TRIGA core	250	1965	3.0E-3	1972
Babcock & Wilcox Lynchburg Pool Reactor Lynchburg, Va.	Pool	1,000	1958	(a)	1974
Dow Chemical Co. Midland, Mich.	U Zr hydride	100	1967	1.0E-3	1972
General Atomic Co. TRIGA-Mk 1 Prototype Reactor LaJolla, Calif.	U Zr hydride	250	1958	6.6E-2	1976
General Atomic Co. Advanced TRIGA-MK 1 Prototype Reactor LaJolla, Calif.	U Zr hydride	1,500	1960	1.61E-1	1976
General Electric Nuclear Test Reactor Pleasanton, Calif.	Light water	100	1957	(b)	1972
Northrop Corp. Labs. (Space Radiation Lab.) Hawthorne, Calif.	U Zr hydride	1,000	1963	9.152	1976
Nuclear Examination Reactor (Rockwell International), Santa Susana, Calif.	Homog.	3	1952	NA	NA
Omaha Veterans Adm. Hospital Omaha, Neb.	U Zr hydride	18	1959	3.0E-1	1972

See footnotes at end of table.

Table 2.5-1. Research reactors (other than DOD or DOE)--continued

Facility	Type	Power kW(t)	Startup date	Airborne Releases Argon-41 Ci	Airborne Releases Time Period
Rhode Island Nuclear Science Center Fort Kearney, RI	Pool	2,000	1964	247	1972
Rockwell International, Canoga Park, Calif.	Homog.	NEG.	1958	NA	NA
U.S. Geological Survey Lab. (Dept. of the Interior), Denver, Colo.	U Zr hydride	1,000	1969	NA	NA
Westinghouse Nuclear Training Center Zion, Ill.		10	1972	NA	NA

NA not available.

NEG. Negligible.

[a] Amount of argon-41 not available but 1.3E-4 Ci of other isotopes were released.

[b] Amount of argon-41 not specified but 27 Ci of noble gases were released.

Emissions

The model facility was assumed to release 200 curies of argon-41 and 100 curies of tritium per year through a 23-meter stack at a velocity of 12.7 m/s (table 2.5-3). These releases were assumed even though they are higher than any shown in table 2.5-1. Thus, they represent a reasonable upper bound for emissions.

Table 2.5-2. Model research reactor facility

Parameter	Value
Type:	Enriched uranium, light-water-cooled and moderated research reactor
Power level:	1 M Watt
Stack:	
Height	23.2 meters
Diameter	1.5 meters
Effluent velocity	12.7 meters/s
Rate of heat emission	0.0

Table 2.5-3. Atmospheric emissions from the model research reactor facility

Radionuclide	Emissions (Ci/y)
Argon-41	200
Tritium	100

2.5.5 Health Impact Assessment of a Model Facility

Table 2.5-4 estimates the annual radiation doses resulting from radioactive emissions from a model research reactor facility. These estimates are for a site in the suburbs of a large Midwestern city (Site B, Appendix A).

Individual fatal cancer risks and committed population fatal cancers are presented in table 2.5-5. The lifetime fatal cancer risk to the maximum exposed group of individuals is estimated to be approximately 1.0E-6. The lifetime fatal cancer risk to the average individual living within 80 kilometers of the model facility is estimated to be 2.4E-9.

The number of fatal cancers committed to the population living in the region near facility is estimated to be 8.2E-5 per year of reactor operations, and the worldwide fatal cancer commitment is estimated to be 6.0E-5 per year of reactor operation.

Table 2.5-4. Annual radiation doses due to radioactive emissions from a model research reactor facility

Organ	Maximum Individual (mrem/y)	Average Individual (mrem/y)	Population (person-rem/y)
Lung	9.1E-2	2.1E-4	5.1E-1
Bone	9.1E-2	2.1E-4	5.1E-1
Kidney	9.1E-2	2.1E-4	5.1E-1
Liver	9.1E-2	2.1E-4	5.1E-1
G.I. Tract	9.1E-2	2.1E-4	5.1E-1
Thyroid	9.1E-2	2.1E-4	5.1E-1
Other soft tissue	9.1E-2	2.1E-4	5.1E-1

Table 2.5-5. Individual lifetime risks and population fatal cancers due to radioactive emissions from a model research reactor facility

Source	Individual lifetime risks Maximum individual	 Average individual	Expected fatal cancers per year of operation (Fatal cancers)
Model facility	1.0E-6	2.4E-9	8.2E-5

Source	Expected total cancers in the worldwide population over the next 100 years (Fatal cancers)
Model facility	6.0E-5

REFERENCES

DOE78 Department of Energy, 1978, Nuclear Reactors Built, Being Built or Planned in the United States as of June 30, 1978, TID-8200-R-38.

NRC77 Letter from J. Kastner, NRC, to W. Mills, EPA, with 59-page enclosure of test, research, and university reactor effluent data, 12/7/77.

2.6 University Reactors

2.6.1 General Description

University reactors are defined as those located at a university and usually operated for the primary purpose of training in the operation and utilization of reactors and for the instruction in reactor theory and performance (DOE78). As of June 30, 1978, there were 54 such reactors (DOE78) operable in the United States (table 2.6-1).

2.6.2 Process Description

University reactors are used primarily as a teaching tool or for basic and applied research. Several types and designs are in use as shown in table 2.6-1. The capable power levels of these reactors is generally low, ranging from negligible to 10 MW.

2.6.3 Emissions of Radionuclides

Due to the relatively low power levels and in some cases intermittent operation, the amounts of radioactive emissions from university reactors are relatively small. University reactors, unlike power reactors, are not required as part of their operating license to report effluent release data to the Nuclear Regulatory Commission (NRC). However, some universities do submit these data to the NRC as a part of their annual operating report and these data are shown in table 2.6-1 (NRC77).

The predominant waste product released by university reactors is argon-41 which is discharged through the facility stack. Argon-41 is produced by neutron activation of argon-40, a naturally occurring component of air. The maximum permissible air concentration (MPC)a for environmental levels of argon-41 is 4.0E-8 microcuries/cc as specified in 10 CFR 20. Most university reactors usually restrict their stack releases to levels below this (MPC)a which is quite conservative since dilution by air in the environment is not taken into account. As a result, the atmospheric concentration of argon-41 in the vicinity of university reactors is generally well below (MPC)a levels.

The differences in reported emission values shown in table 2.6-1 are probably due to variations in operating time, power levels, and control systems. Universities with large student demand for reactor time obviously operate their reactors longer during a given year than do universities with smaller programs. Specific data concerning operating parameters of the reactors listed in table 2.6-1 are not compiled.

Table 2.6-1. University reactors (other than DOD or DOE)

Facility	Type	Power kW(t)	Startup date	Airborne Releases Argon-41 (Ci)	Airborne Releases Time Period
Brigham Young University Provo, Utah	Homog.	NEG.	1967	NA	NA
California State Polytechnic San Luis Obispo, Calif.	Homog. solid	NEG.	1973	NA	NA
Catholic Univ. of America Washington, D.C.	Homog. solid	NEG.	1957	NA	NA
Columbia Univ. New York, N.Y.	U Zr hydride	250	1977	NA	NA
Cornell Univ. Ithaca, N.Y.	U Zr hydride	100	1962	1.16	1972
Cornell Univ. Zero Power Reactor Ithaca, N.Y.	Tank	NEG.	1962	NA	NA
Georgia Institute of Technology Atlanta, Ga.	Homog. solid	NEG.	1957	NA	NA
Georgia Tech. Research Reactor Atlanta, Ga	Heavy water	10,000	1964	466.87	1976

See footnotes at end of table.

Table 2.6-1. University reactors (other than DOD or DOE)--continued

Facility	Type	Power kW(t)	Startup date	Airborne Releases Argon-41 (Ci)	Airborne Releases Time Period
Idaho State Univ. Pocatello, Idaho	Homog. solid	NEG.	1967	NA	NA
Iowa State Univ. Ames, Iowa	Graphite water	10	1959	0.4	1972
Kansas State University Manhattan, Kan.	U Zr hydride	250	1962	6.0E-4	1972
Manhattan College New York, N.Y.	Tank	NEG.	1964	NA	NA
Massachusetts Institute of Tech. Cambridge, Mass.	Heavy water reflected	5,000	1958	[a]7795.0	7/76-6/77
Memphis State Memphis, Tenn.	Homog. solid	NEG.	1977	NA	NA
Michigan State University East Lansing, Mich.	U Zr hydride	250	1969	1.1E-3	1976
North Carolina State University Raleigh, N.C.	Pool	1,000	1972	NA	NA
Nuclear Science Center Reactor Texas A&M Univ., College Station, Texas	Pool TRIGA core	1,000	1961	3.35	1976
Ohio State Univ. Columbus, Ohio	Pool	10	1961	3.4E-2	1972
Oregon State University Corvallis, Ore.	Homog. solid	NEG.	1958	NA	NA

See footnotes at the end of the table.

Table 2.6-1. University reactors (other than DOD or DOE)--continued

Facility	Type	Power kW(t)	Startup date	Airborne Releases Argon-41 (Ci)	Airborne Releases Time Period
Oregon State Corvallis, Ore.	U Zr hydride	1,000	1967	20.06	7/76-6/77
Penn. State TRIGA Reactor, Penna. State University, Univ. Park, Pa.	Pool TRIGA core	1,000	1965	1	1972
Purdue University West LaFayette, Ind.	Pool	10	1962	NA	NA
Reed College Portland, Ore.	U Zr hydride	250	1968	NA	NA
State Univ. of New York (Western New York Nuclear Research Center, Inc.) Buffalo, N.Y.	Pool	2,000	1961	6.9	1972
Texas A&M Univ. College Station, Texas	Homog. solid	NEG.	1957	NA	NA
Tuskegee Institute Tuskegee, Ala.	Homog. solid	NEG.	1957	NA	NA
Univ. of Arizona Tucson, Ariz.	U Zr hydride	250	1958	NA	7/76-5/77
Univ. of Calif. Berkeley, Calif.	U Zr hydride	1,000	1966	32	1972
Univ. of Calif. Santa Barbara, Calif.	Homog.	NEG.		None	

See footnotes at the end of the table.

Table 2.6-1. University reactors (other than DOD or DOE)--continued

Facility	Type	Power kW(t)	Startup date	Airborne Releases Argon-41 (Ci)	Airborne Releases Time Period
Univ. of Calif. Irvine, Calif.	U Ar hydride	250	1969	4.2E-2	7/77-6/77
Univ. of Calif at Los Angeles School of Engineering & Applied Science Los Angeles, Calif.	Graphite water	100	1960	33.0	1976
Univ. of Delaware Newark, Dela.	Homog. solid	NEG.	1958	NA	NA
Univ. of Florida Gainesville, Fla.	Gaphite/ water	100	1959	5.03	9/75-8/76
University of Illinois Urbana, Ill.	U Zr hydride	10	1971	NA	NA
University of Illinois Urbana Champaign, Ill.	U Zr hydride	1,500	1960	[b]2.60	1976
Univ. of Kansas Lawrence, Kansas	Pool	10	1961	NA	7/76-6/77
Univ. of Lowell Lowell, Mass.	Pool	1,000		NA	NA
Univ. of Md. College Park, Md.	Tank	350	1960	NA	NA
Univ. of Michigan (Ford Nuclear Reactor) Ann Arbor, Mich.	Pool	2,000	1957	(c)	1976

See footnotes at the end of the table.

Table 2.6-1. University reactors (other than DOD or DOE)--continued

Facility	Type	Power kW(t)	Startup date	Airborne Releases Argon-41 (Ci)	Time Period
Univ. of Missouri Columbia, Mo.	Tank	10,000	1966	[d]2,406	6/76-7/77
Univ. of Missouri at Rolla Rolla, Mo.	Pool	200	1961	(e)	1976
Univ. of New Mexico Albuquerque, N.M.	Homog. solid	NEG.	1957	NA	NA
Univ. of Oklahoma Norman, Okla.	Homog. solid, Pool	NEG.	1958	NA	NA
Univ. of Texas Austin, Texas	U Zr hydride	250	1963	NA	1976
Univ. of Utah Salt Lake City, Utah	U Zr hydride	250	1975	NA	NA
Univ. of Virginia Charlottesville, Va.		NEG.		[f]0.442	1976
Univ. of Virginia Charlottesville, Va.	Pool	2,000	1960	NA	NA
Univ. of Washington Seattle, Wash.	Graphite/ water	100	1961	13.0	1976
University of Wisconsin Madison, Wisc.	Pool TRIGA core	1,000	1960	1.77	7/77-6/76

See footnotes at end of the table.

Table 2.6-1. University reactors (other than DOD or DOE)--continued

Facility	Type	Power kW(t)	Startup date	Airborne Releases Argon-41 (Ci)	Airborne Releases Time Period
Virginia Polytechnic Institute Blacksburg, Va.	Graphite/ water	100	1959	180.14	1976
Washington State Univ. Pullman, Wash.	Pool TRIGA core	1,000	1961	7.67	7/76-6/77
Worcester Polytechnic Institute Worcester, Mass.	Pool	10	1959	NA	NA

NA not available.
NEG. Negligible.
[a]Also 7.04 Ci of tritium.
[b]Also 1.0E-3 Ci of tritium.
[c]A total of 107.9 Ci of unspecified radioactivity.
[d]Also 4.8 Ci of tritium.
[e]Also 0.115 of unspecified radioactivity.
[f]Also 0.013 Ci of krypton-85.

2.6.4 Model Facility

In order to estimate population and individual radiation doses, a model facility (table 2.6-2) has been developed.

Emissions

Atmospheric emissions from the model facility are reported in table 2.6-3. The argon-41 releases are somewhat high when compared with the majority of the source terms given in table 2.6-1; however, two reported values are greater than 1,000. The assumed annual release for tritium is higher than any shown in table 2.6-1. The argon-41 release term assumed here, therefore, represents a reasonable upper bound for this source category.

Table 2.6-2. Model university reactor facility

Parameter	Value
Type:	Enriched uranium, heavy-water moderated and cooled university reactor
Power level:	10,000 kW(t)
Stack:	
Height	23.2 meters
Diameter	1.5 meters
Effluent velocity	12.7 meters/s
Rate of heat emission	0.0

Table 2.6-3. Atmospheric emissions of radionuclides from the model university reactor facility

Radionuclide	Emissions (Ci/y)
Argon-41	1,000
Tritium	100

2.6.5 Health Impact Assessment of A Model University Reactor Facility

Table 2.6-4 presents estimates of annual radiation doses resulting from radioactive emissions from a model university reactor facility. These estimates are for a site located in the suburbs of a large Midwestern city (Site B, Appendix A).

Individual fatal cancer risks and committed population fatal cancers are presented in table 2.6-5. The lifetime fatal cancer risk to the maximum exposed group of individuals is estimated to be approximately 4.9E-6. The lifetime fatal cancer risk to the average individual living within 80 kilometers of the model facility is estimated to be 1.1E-8.

The number of fatal cancers committed to the population living in the region around the facility is estimated to be 3.9E-4 per year of reactor operation. The estimated number of fatal cancers committed to the world population is 6.0E-5 per year of reactor operation.

Table 2.6-4. Annual radiation doses due to radioactive emissions from a model university reactor facility

Organ	Maximum individual (mrem/y)	Average individual (mrem/y)	Population (person-rem/y)
Lung	4.3E-1	9.7E-4	2.4
Bone	4.3E-1	9.7E-4	2.4
Kidney	4.3E-1	9.7E-4	2.4
Liver	4.3E-1	9.7E-4	2.4
G.I. Tract	4.3E-1	9.7E-4	2.4
Thyroid	4.3E-1	9.7E-4	2.4
Other soft tissue	4.3E-1	9.7E-4	2.4

Table 2.6-5. Individual lifetime risks and population fatal cancers due to radioactive emissions from a model university reactor facility

Source	Individual lifetime risks Maximum individual	 Average individual	Expected fatal cancers per year of operation (Fatal cancers)
Model facility	4.9E-6	1.1E-8	3.9E-4

Source	Expected total cancers in the worldwide population over the next 100 years (Fatal cancers)
Model facility	6.0E-5

REFERENCES

DOE78 Department of Energy, 1978, Nuclear Reactors Built, Being Built or Planned in the United States as of June 30, 1978, TID-8200-R-38.

NRC77 Letter from J. Kastner, NRC, to W. Mills, EPA, with 59-page enclosure of test, research, and university reactor effluent data. 12/7/77.

2.7 Shallow Land Burial of Low-Level Radioactive Wastes

2.7.1 General Description

Current practice in the United States is to dispose of solid low-level radioactive waste by shallow land burial. These wastes are generated primarily by the nuclear fuel cycle, nuclear medicine facilities, commercial manufacturers, universities, and private and governmental research institutions.

Table 2.7-1 lists the six commercial low-level waste burial facilities and five major active sites operated by Department of Energy (Ad78). Three of the commercial facilities are currently inoperative. Maxey Flats, Ky, and West Valley, N.Y., are closed indefinitely and the responsibility for perpetual care is expected to be shifted to the respective States. The Sheffield, Illinois, site is awaiting approval from NRC and the State of Illinoisto expand its storage capacity.

2.7.2 Process Description

In general the wastes are placed as received in a trench excavated in the existing till or soil and the material thus removed is used to cover the wastes once the trench is filled to capacity. This overburden is sometimes compacted and usually mounded to promote surface water runoff. The dimensions of trenches and their proximity to adjacent trenches vary within and between sites.

The character, source, and amount of specific radionuclides contained in the waste differ somewhat from site to site. However, the operational characteristics and the compositions of the wastes are generally similar.

2.7.3 Emissions of Radionuclides

Emissions from Normal Waste Handling Operations

The routine procedures of transporting, packaging, and burying the waste at the low-level radioactive waste burial site may result in the release and subsequent dispersion of radioactive material into the atmosphere. All radionuclides contained in the waste are potential sources.

Table 2.7-1. Low-level radioactive waste burial sites, United States, 1979

Location	Operator	Licensed by	Originally licensed by (year)	Status	Commercial capacity (m^3)	Climate
Hanford, Washington	DOE	-	-	-	-	Semi-Arid
Richland, Washington	NECO	State of Wash. and NRC	AEC (1965)	Open	9.0E+5	Semi-Arid
Beatty, Nevada	NECO	State of Nev.	AEC (1962)	Open	7.0E+5	Arid
Inel, Idaho	DOE	-	-	-	-	Semi-Arid
Los Alamos, New Mexico	DOE	-	-	-	-	Semi-Arid
Sheffield, Illinois	NECO	State of Ill. and NRC	AEC (1967)	Closed	2.0E+5	Humid
Morehead, Kentucky (Maxey Flats)	NECO	State of Ky.	Ky. (1962)	Closed	3.0E+5	Humid
Oak Ridge, Tennessee	DOE	-	-	-	-	Humid
Savannah River, S.C.	DOE	-	-	-	-	Humid
Barnwell, S.C.[a]	CSI	State of S.C. and NRC[b]	S.C. (1971)	Open	2.0E+6	Humid
West Valley, N.Y.	NFS	State of N. Y.	N.Y. (1963)	Closed	2.0E+5	Humid

AEC Atomic Energy Commission.
CSI Chem-Nuclear Systems, Inc.
NECO Nuclear Engineering Company, Inc.
DOE Department of Energy.
NRC Nuclear Regulatory Commission.
NFS Nuclear Fuel Services.

[a] Present State licensing conditions limit yearly burials of waste to 7.0 E+4 m^3 of waste per year.

[b] NRC licenses only special nuclear material.

Resuspension of Contaminated Soil

The surface soil of a waste burial site can become contaminated with radionuclides during normal operations. The contaminated soil particles may then be resuspended and transported offsite. Those particles which are respirable may contribute a dose to the respiratory tract of persons in the general population. These contaminants may also expose persons in the surrounding population through uptake and transport in other pathways, e.g., uptake into food crops. All radionuclides routinely received at the waste burial site could be resuspended.

Emissions from Trenches

Long-lived radionuclides either existing in a gaseous state or those forming gaseous compounds in the trench gas may migrate through the trench cover and be dispersed to areas off the waste burial sites and result in doses to the public. Radionuclides which may be released are krypton-85, radon-222, carbon-14, and tritium.

Emissions from Leachate Evaporator

Waste trenches excavated in soil (till) which is impervious to water in its undisturbed state retain most surface water entering through a fractured trench cap. Thus, trenches in humid (high precipitation) areas may overflow if the water collected (leachate) is not periodically removed. At the Maxey Flats site, after pumping and storage, the leachate is concentrated to solids by an evaporator. The vaporized liquid containing volatized radionuclides is released to the atmosphere.

2.7.4 Typical Facility

To estimate airborne emissions and the health impact of low-level radioactive waste burial sites, the Maxey Flats site was chosen because it is considered to be representative of sites located in humid areas and most of the data needed for calculations are available. However, since there were no data on gaseous releases from trench caps, West Valley data was used because that site is similar in geology, meteorology, and in the type of buried wastes. In addition both sites use leachate evaporators to avoid trench overflow. The data on emissions from leachate evaporators are presented separately to assess its impact.

Typical facility parameters are given in table 2.7-2. A comprehensive inventory of the radioactivity buried at Maxey Flats

Table 2.7-2. Typical low-level radioactive waste burial site

Parameter	Value
Area	1.3 km^2
Radionuclide inventory	(See table 2.7-3)
Annual rainfall	125 cm/y
Evaporator stack height	10 meters
Regional population	387,961

Table 2.7-3. Annual atmospheric emissions of radionuclides and inventory of principal radionuclides at a typical low-level radioactive waste burial site

		Emissions (Ci/y)		
Radionuclide	Total in place at site (Ci)	Normal Operations	Trenches	Leachate Evaporator
Tritium	5.4E+5	5.4E-03	80.0	1.9E+4
Carbon-14	3.6E+4	3.6E-04	5.0	5.6E-3
Iron-55	1.2E+2	1.2E-06	-	-
Cobalt-58	7.0E-2	7.0E-10	-	7.8E-4
Cobalt-60	8.1E+4	8.1E-04	-	2.4E-3
Strontium-90	1.5E+4	1.5E-04	-	1.9E-3
Technetium-99	9.3	9.3E-08	-	-
Iodine-129	1.1E-2	1.1E-10	-	-
Cesium-134	3.9E-1	3.9E-09	-	8.8E-4
Cesium-137	2.5E+4	2.5E-04	-	3.5E-2
Radium-226	4.9E+3	4.9E-05	-	-
Uranium-233	9.5E+2	9.5E-06	-	-
Uranium-235	3.5E-1	3.5E-09	-	-
Uranium-238	8.5E+1	8.5E-07	-	-
Plutonium-238	4.3E+4	4.3E-04	-	4.3E-4
Plutonium-239	[a]3.8E+3	3.8E-05	-	8.1E-6

[a] Includes 2.1E+02 of plutonium of uncertain isotopic origin.

was compiled from radioactivity shipment records by Gat et al. (GA75) in 1975 and was recently updated by Prairie (PR76). The inventory of the principal radionuclides is shown in table 2.7-3.

Emissions from Normal Waste Handling Operations

Adams has calculated emissions for major radionuclides from normal waste handling operations at Maxey Flats (Ad78). The same methodology has been used in calculating the releases in table 2.7-3, but the nuclide inventory is from Prairie (Pr76).

Emissions from Trenches

The Office of Radiation Programs is currently studying gaseous releases from trenches at four low-level burial sites. The trench pathway was identified by Davis et al. (Da76) in studies at West Valley. The only data currently available are carbon-14 and tritium as methane in the forms of $^{14}CH_4$ and $^{3}HCH_3$, respectively. These data have been reported by Lu, Matuszek, et al. (Lu78, Ma78) and are given in table 2.7-3. The ORP studies will identify the magnitude of carbon-14, tritium, krypton-85, and radon-222 releases at four sites as well. Carbon-14 and tritium releases at Maxey Flats are not expected to differ greatly from those at West Valley because of the similarities of the two sites.

Emissions from Leachate Evaporator

This pathway has been studied by Blanchard, et al. (Bl78). The average annual release is calculated by using the product of the average discharge rates and the annual operating time for the evaporator (table 2.7-3).

2.7.5 Health Impact Assessment of a Typical Low-Level Radioactive Waste Burial Site

Table 2.7-4 estimates annual radiation doses resulting from radioactive emissions from the typical low-level radioactive waste burial site. The estimates are for a rural, low population density site in Fleming County, Kentucky.

Table 2.7-5 estimates the individual lifetime risks and fatal cancers to the population from these doses. The lifetime cancer risk to the highest exposed group of individuals is approximately 8.5E-5 resulting primarily from emissions from the leachate evaporator. The lifetime cancer risks to the average individual in the region is estimated to be 5.0E-8.

Table 2.7-4. Annual radiation doses from radioactive emissions from a typical low-level radioactive waste burial site

Organ	Leachate evaporator		
	Maximum individual (mrem/y)	Average individual (mrem/y)	Population (person-rem/y)
Lung	1.8E+1	1.1E-2	4.1
Bone	2.5E+1	1.2E-2	4.6
Kidney	1.9E+1	1.1E-2	4.2
Liver	1.9E+1	1.1E-2	4.2
Thyroid	1.9E+1	1.1E-2	4.2
G.I. Tract	1.7E+1	1.1E-2	4.1
Other soft tissue	1.9E+1	1.1E-2	4.2

Organ	Area sources		
	Maximum individual (mrem/y)	Average individual (mrem/y)	Population (person-rem/y)
Lung	6.8E-1	4.9E-4	1.9E-1
Bone	5.4	2.4E-2	9.3E-1
Kidney	9.6	6.4E-4	2.5E-1
Liver	1.1	7.2E-4	2.8E-1
Thyroid	5.5E-1	4.9E-4	1.9E-1
G.I. Tract	6.4E-1	7.5E-4	2.9E-1
Other soft tissue	8.9E-1	9.5E-4	3.7E-1

Organ	Total		
	Maximum individual (mrem/y)	Average individual (mrem/y)	Population (person-rem/y)
Lung	1.9E+1	1.1E-2	4.3
Bone	3.0E+1	1.4E-2	5.5
Kidney	2.0E+1	1.2E-2	4.5
Liver	2.0E+1	1.2E-2	4.5
Thyroid	2.0E+1	1.1E-2	4.4
G.I. Tract	1.8E+1	1.2E-2	4.4
Other soft tissue	2.0E+1	1.2E-2	4.6

The number of fatal cancers per year of site operation is estimated to be 7.5E-4 to the population in the region around the site and 3.3E-2 to the world population. These result primarily from emissions from the leachate evaporator. The number of fatal cancers committed to the world population is the impact over a 100-year period of time from emissions of carbon-14 and tritium during a one-year period.

Health impacts were assessed for a plant with an assumed operating lifetime of 25 years.

Table 2.7-5. Individual lifetime risks and population fatal cancers due to radioactive emissions from a typical low-level radioactive waste burial site

Area	Individual lifetime risks[a] Maximum individual	Individual lifetime risks[a] Average individual	Expected fatal cancers per year of operation (Fatal cancers)
Leachate evaporator	8.0E-5	4.5E-8	6.8E-4
Area sources	6.8E-6	4.3E-9	6.6E-5
Total	8.5E-5	5.0E-8	7.5E-4

Radionuclide	Expected total cancers in the worldwide population over the next 100 years Leachate evaporator	Expected total cancers in the worldwide population over the next 100 years Area sources
Tritium	1.2E-2	4.8E-5
Carbon-14	2.5E-5	2.1E-2
Total[b]	1.2E-2	2.1E-2

---

[a] Risks over the lifetime of an individual for an exposure to the dose rates shown in table 2.7-5 for a 25-year period.

[b] Impact over 100-year period from one-year release of carbon-14 and tritium.

REFERENCES

Ad78 Adams J. A., and Rogers V. L., 1978, A Classification System for Radioactive Waste Disposal--What Waste Gas Where?, NUREG-0456, Nuclear Regulatory Commission, Washington, D.C.

Bl78 Blanchard R. L., Montgomery D. M., Kolde H. E., and Meyer G. L., 1978, Supplementary Radiological Measurements at the Maxey Flats Radioactive Waste Burial Site--1976-1977, EPA-520/5-78-011, Environmental Protection Agency, Eastern Environmental Radiation Facility, Montgomery, Ala.

Da76 Davis J. F., Fakundiny R. H., Pferd J. IV, 1976, Evaluation of Radionuclide Pathways at a Shallow Low-Level Radioactive Waste Burial Site in Western New York, presented at the National Meeting of the Geological Society of America, November 10, 1976.

Ga75 Gat U., Thomas J. D., and Clark D. T., 1975, Radioactive Waste Inventory at the Maxey Flats Nuclear Waste Burial Site, Health Physics, 30, 281.

Lu78 Lu A. H. and Matuszek J. M., 1978, Transport through a Trench Cover of Gaseous Tritiated Compounds from Buried Radioactive Wastes, presented at the International Symposium on the Behavior of Tritium in the Environment, San Francisco, Calif., October 16-20, 1978.

Ma78 Matuszek J. M., Wahlen M. U., Kunz, C. O., and Hutchinson J. H., 1978. Is the Removal of ^{14}C from Nuclear Reactor Effluents a Plausible Regulatory Practice?, presented at a conference on Effluent and Environmental Radiation Surveillance, Atlanta, Ga.,July 9-14, 1978.

Mo77 Montgomery D. E., Kolde H. E., and Blanchard R. L., 1977, Radiological Measurements at the Maxey Flats Radioactive Waste Burial Site--1974-1975, EPA-520/5-76/020, January 1977, Eastern Environmental Radiation Facility, Montgomery, Ala.

Pr76 Richard Prairie, Personal Communication.

2.8 Plutonium Fuel Fabrication Facilities

2.8.1 General Description

The NRC currently licenses seven facilities within their Plutonium Fuel and Fabrication (Special Nuclear Materials) category. These facilities and their locations are shown in table 2.8-1. The missions of these facilities range from laboratory-scale plutonium fuel research to fabrication of light-water plutonium recycle reactor fuel. In general, these facilities use processes which are still under development with regard to the technology and production of plutonium or plutonium-uranium mixed oxide fuel elements. The overall objective is to recover plutonium and uranium from depleted uranium process residues and fabricate these isotopes as oxides into fuel pellets.

2.8.2 Process Description

Fuel pellets are fabricated from plutonium oxide and natural uranium oxide powders which are received as feed materials (ORNL75). Another possible feed material is aqueous plutonium nitrate solution which is converted to plutonium oxide powder. Scrap metal alloys bearing plutonium may also be used.

Plutonium-bearing solids are mixed with nitric acid to dissolve the plutonium. Impure plutonium nitrate is then pumped through ion exchange resin beds for selective sorption of the plutonium. The plutonium on the resin column is eluted and transferred to the product evaporator. The resin is reconditioned and the scrap recovery cycle is repeated.

Plutonium Oxide Conversion Process

Plutonium is precipitated either as the peroxide or oxalate which produces a filter cake. The filter cake is calcined batch-wise in a muffle furnace to produce PuO_2. The oxide is milled, screened, and packaged for transfer to the fabrication process.

Mixed Oxide Preparation

Mixed oxides in plutonium and uranium are prepared by combining UO_2 with PuO_2 in a V-blender.

Pellet Production

The mixed oxide blend is pressed, sintered, and dry-centerless-ground into cylindrical pellets.

Fuel Rod Production

Pellets are put into fuel rods. Their subassembly is then decontaminated and welded into a final fuel rod assembly.

Control Technology

Radiological control systems consist of single, double, or triple-stage HEPA filtration on process stream ventilation systems before there is release to the atmosphere.

2.8.3 Emissions of Radionuclides

Because plutonium is valuable, great care is taken to avoid losing any of this material; filters from the HEPA systems are returned to the scrap recovery portion of the process.

Differences in reporting by facilities preclude listing all available data on emissions of radionuclides in one table. Some facilities report gross alpha emissions only, while others provide specific isotope emissions. Table 2.11-2 shows data which are available on a comparable basis.

2.8.4. Model Plutonium Fuel Fabrication Facility

A model facility (table 2.8-3) was developed for assessing the impact of airborne emissions from plutonium fuel fabrication plants. The model parameters and source terms were taken from Groenier, et al. (ORNL75) based on their Case 1 (base case, current practice) data. The source terms used for the model are shown in table 2.8-4.

Table 2.8-1. Plutonium fuel fabrication facilities (NRC79)

Facility	Location
Babcock and Wilcox	Parks Township, Pa.
Battelle Memorial Institute	West Jefferson, Ohio
Battelle P. N. Laboratories	Richland, Wash.
Exxon Nuclear Co., Inc.	Richland, Wash.
General Electric Co.	Vallecitos, Calif.
Westinghouse Electric Corp.	Cheswick, Pa.
Westinghouse Electric Corp.	Waltz Mill, Pa.

Table 2.8-2. Atmospheric emissions of radionuclides from currently licensed plutonium fuel fabrication facilities[a]

Facility	Release interval	Plutonium emissions[b] (Ci)
Exxon Nuclear	1976	4.8E-3
Battelle (Memorial)	1976	6.3E-7
Westinghouse (Cheswick)	7-12/78	<1.4E-4

[a] Source: Facility reports to NRC.

[b] Contain plutonium-238, plutonium-239, plutonium-240, plutonium-241, and plutonium-242.

Table 2.8-3. Model plutonium fuel fabrication facility

Parameter	Value
Type of facility	Plutonium fuel fabrication
Annual capacity	300 MT
Stack	10-meter fixed stack height with no plume rise

Table 2.8-4. Atmospheric emissions of radionuclides from the model plutonium fuel fabrication facility

Radionuclide	Emissions (Ci/y)
Uranium-234	1.0E-09
Uranium-235	3.3E-11
Uranium-238	7.1E-10
Plutonium-238	4.9E-05
Plutonium-239	2.6E-06
Plutonium-240	5.8E-06
Plutonium-241	1.3E-06
Plutonium-242	3.6E-08
Americium-241	4.3E-06

2.8.5. Health Impact Assessment for Model Plutonium Fuel Fabrication Facility

Annual radiation doses due to airborne radioactive emissions from the model plutonium fuel fabrication facility are presented in table 2.8-5. These estimates are for a facility located in the near suburbs of a large Midwestern city (Site B with rural food source fractions, Appendix A). The maximum individual dose equivalent rate occurred 503 meters downwind of the site.

Table 2.8-6 estimates the individual lifetime fatal cancer risks and committed fatal cancers to the regional population. The lifetime fatal cancer risk to the highest exposed group of individuals is approximately 1.1E-6. The lifetime fatal cancer risk to the average individual in the region is estimated to be 6E-10.

The estimated number of fatal cancers per year of site operation to the regional population is estimated to be 2.1E-5.

Table 2.8-5. Annual radiation doses from radioactive emissions from the model plutonium fuel fabrication facility

Organ	Maximum individual (mrem/y)	Average individual (mrem/y)	Population (person-rem/y)
Lung	1.7E-2	9.6E-6	2.4E-2
Bone	4.0E-1	2.2E-4	5.6E-1
Kidney	4.5E-2	2.5E-6	6.3E-2
Liver	6.1E-2	3.3E-6	8.3E-2
Thyroid	1.1E-2	6.4E-7	1.6E-2
G.I. Tract	9.5E-4	4.8E-7	1.2E-3
Other soft tissue	1.1E-2	6.4E-6	1.6E-2

Table 2.8-6. Individual lifetime risks and population fatal cancers due to radioactive emissions from the model plutonium fuel fabrication facility

Source	Individual lifetime risks Maximum individual	 Average individual	Expected fatal cancers per year of operation (Fatal cancers)
Model facility	1.0E-6	5.6E-10	2.0E-5

REFERENCES

NRC79 U.S. Nuclear Regulatory Commission, 1979, Program Summary Report, NUREG 0380, Vol. 3, No. 3, March 16, 1979.

ORNL75 Oak Ridge National Laboratory, 1975, Correlation of Radioactive Waste Treatment Costs and the Environmental Impact of Waste Effluents in the Nuclear Fuel Cycle for Use in Establishing "as low as Practicable" Guides--Fabrication of Light-Water Reactor Fuels Containing Plutonium, ORNL-TM-4904, May 1975.

2.9 Industrial Users and Other Categories

2.9.1 General Description

Table 2.9-1 lists the categories of industrial licenses and the number of licenses issued in each category by the Nuclear Regulatory Commission (NRC78). Information from the NRC Office of Standards Development (TC78) indicate that almost all of the radiation sources covered by these licenses are sealed (encapsulated) sources, i.e., the industrial application of the radionuclide uses electromagnetic radiation from a sealed source.

The remaining sources can emit radionuclides into the air. These include facilities which make static eliminators (nonsealed alpha or beta emitters), self-illuminating devices (such as tritium activated signs, watch dials), and facilities engaged in research and development.

2.9.2. Process Description

The industrial uses of radionuclides are many; the following are examples.

Radioisotope Gauges

A sealed radionuclide source is placed on one side of a production line with a radiation detection device on the other side. These gauges automatically measure and control the thickness of steel sheet, paper, tire cords, textiles, plastics and other materials. They may also control sugar, fat, and meat content in food packaging; measure soil density and moisture content; and measure snow density for water run-off forecasting.

These devices do not normally emit radionuclides into the air.

Static Eliminators

Electroplated alpha or beta sources are placed in close proximity to production components prone to the build-up of static electricity (such as moving conveyor belts). The ionizing characteristics of the radiation reduce static charges. They are used in production systems where static charges may degrade product quality or pose safety hazards (inflammable or explosive environments).

These devices do not normally emit radionuclides into the air.

Table 2.9-1. NRC Industrial License Categories

Program area	Licenses	Licensees
Industrial:		
Well logging	73	67
Other measuring systems*	2,373	1,985
Manufacturing, distribution and service-broad	60	47
Manufacturing, distribution and service-other	258	214
General license distribution	69	11
Exempt quantities	46	28
Exempt watches	63	55
Other exempt distribution	104	54
Nuclear laundry	5	5
Leak test	31	28
Waste disposal (Burial)	3	3
Waste disposal (other)	10	7
Radiography-single location	147	132
Radiography-multiple location	214	177
Power source	1	1
Irradiator (<10,000 curies)	177	41
Irradiator (>10,000 curies)	54	11
Research & development broad	83	65
Research & development other	425	343
Total	4,196	3,274

*Other measuring systems include thickness gauges, level indicators, densitometers, etc.

Radiographic Devices

These devices use shielded sealed gamma sources as portable radiography units for nondestructive testing. They are popular as field quality control and inspection devices in such areas as pipeline construction, building, and aircraft inspection to detect flaws, voids, and other defects in materials and equipment. There are no routine emissions of radionuclides into air.

Manufacturing and Research and Development Facilities

These facilities use radionuclides, sometimes in large quantities, in manufacturing such product as self illuminating watch dials using tritium, smoke detectors using americium-241, and static illuminators. These are also many facilities which radionuclides in research and development processes.

Emission data from this category of sources is not now available.

2.9.3 Emission of Radionuclides

No data are available.

2.9.4. Model Facility

No model was developed, pending availability of release data.

2.9.5. Health Impact Assessment

An assessment will be made when emission data are available.

2.9.6. Other Licensed Categories

Other NRC license categories may be potential significant sources of airborne emissions of radionuclides. These categories include the academic license categories (university laboratories) and byproduct material licenses for facilities performed in research which are not included in the industrial license category by NRC. Radionuclide emission data are not now available. Information on these categories has been requested from NRC and an analysis of them will be included in future reports.

REFERENCES

NRC78 Memorandum from NRC, Office of Standards Development to Robert S. Callis, EPA, dated November 28, 1978.

TC78 Telephone conversation with Lewis Battist, NRC, Environmental Protection Standards Branch, November 27, 1978.

CHAPTER 3

SOURCES OF EMISSIONS FROM DEPARTMENT OF ENERGY FACILITIES

Department of Energy (DOE) facilities support the development of weapons in response to Department of Defense requirements. DOE is also responsible for research programs on the biomedical, environmental, physical and safety aspects of nuclear energy. Twenty-seven prime contractor sites are discussed in this section. Data presented here have been obtained from reports by the contractor to DOE during 1977.

The assessments for DOE facilities therefore were made on a site-by-site basis. The diversity of operations at the different sites makes it difficult to assess DOE facilities on a generic basis. In general, the annual doses reported are 50-year dose commitments, i.e., the dose delivered to an organ over a 50-year period from one year's exposure. For most radionuclides, however, the dose commitment is delivered in the first year. Several sites report 70-year dose commitments. These are appropriately annotated in the text. Additional information about the method by which the reported doses were calculated can be found in the references to each section. The maximum exposed individual at DOE sites represents either a hypothetical individual at the site boundary or the nearest resident or group of individuals.

For the purpose of consistency of notation in the summary table S-2, the 50-year or 70-year dose commitments are listed under the annual dose rates; these are appropriately notated in the table and in the individual site sections.

3.1 Hanford Site

3.1.1 General Description

The Hanford site was established in 1943 to produce plutonium for nuclear weapons. Today, operations at the facilities on the Hanford site include plutonium production, reactor fuel reprocessing, waste management, reactor fuel manufacturing and research and development.

Hanford site consists of 145,000 hectares of Government-owned land in south-central Washington State. The site is about 320 kilometers east of Portland, Oregon, 270 kilometers southeast of Seattle, Washington, and 200 kilometers southwest of Spokane, Washington. The Columbia River flows through the northern area and along the eastern boundary.

The Hanford environs are categorized as a desert plain. The rural, agricultural region is sparsely populated--only 250,000 people live within 80 kilometers of the site. The nearest population center includes the tri-city area of Richland, Pasco, and Kenewich which have a combined population of 80,000. These cities are southeast and downriver of the Hanford Site.

3.1.2 Process Description

100 Area

The 100 Area of the Hanford site, located to the north along the Columbia River, is the location of the historic plutonium production reactors. Eight reactors are deactivated and only the N Reactor remains operational--the principal source of radioactive atmospheric releases in the area.

The exhaust air from the reactor building undergoes absolute and activated charcoal filtration before being released. The stack gases are monitored for gamma radiation and continuously sampled. Radioactive atmospheric releases originating from the reactor heat exchangers are not treated. However, the exhaust air is sampled continuously. Argon-41 constitutes 95 percent of the radioactivity released into the atmosphere from the 100 Area.

200 AREA

The 200 Area in the center of the reservation is the most isolated area and the most removed from surface and subsurface water. This area contains the fuel processing and waste-management facilities and is subdivided into the 200 East and 200 West Areas.

200 EAST AREA

The major facilities located within the 200 East Area are the Purex plant, B plant, Chemical Processing Plant, Critical Mass Laboratory and the Tank Farm Facility. The Purex plant is maintained in a standby status and is used only for plutonium recovery when required by production from the N Reactor. The B plant facility is used for waste fractionization, encapsulation and storage. Cesium-137 and strontium-90 are removed from process wastes, then doubly encapsulated and stored in retrievable water-cooled storage. The Chemical Processing Plant is presently shut down. At the Critical Mass Laboratory, research concerning criticality safety of plutonium is conducted.

High-level radioactive wastes, produced as a result of the chemical reprocessing of irradiated Hanford reactor fuels, are stored in large underground tanks. Operations at the 200 East Area Tank Farm facilities involve evaporating nonboiling waste to salt cake. There are three evaporators in operation, two of which are permanently installed in tanks for in-tank solidification.

Most of the radioactive atmospheric effluents from the 200 East Area facilities undergo some form of HEPA filtration. Effluents are either monitored or regularly sampled prior to release. The 200 East Area facilities release small quantities of gross alpha activity, assumed to be plutonium-239, and 0.12 curies unidentified beta and gamma activity from mixed fission products, which were assumed to be strontium-90.

200 WEST AREA

The U plant, Redox plant, T plant, Z plant and the Tank Farm are the major facilities located within the 200 West Area. The U plant is no longer used to recover uranium. However, the adjacent uranium oxide plant produces powdered UO_3 by calcining uranyl nitrate hexahydrate ($UO_2(NO_3)_2 6H_2O$). The Redox fuel processing plant is also shut down, only its analytical laboratory remains in operation. The T plant is now used for various decontamination projects, equipment repair, and plutonium storage. The Z plant has only a plutonium laboratory which operates when required by the N Reactor. Operations include processing and preparation of plutonium materials for the N Reactor.

The Tank Farms and solidification facilities in the area consist of 149 underground waste storage tanks that have capacities for 190,000 to 3,800,000 liters of high-level liquid radioactive waste, and two evaporators for the reduction of liquid waste to a solid salt cake.

The radioactive atmospheric emissions from the 200 West Area, like those from the East Area, undergo one or more stages of HEPA filtration before being released. Effluent release points are monitored by routine sampling. The 200 West Area emits small amounts gross alpha activity, assumed to be plutonium-239, and millicurie quantities of unidentified beta and gamma radioactivity, assumed to be strontium-90.

300 AREA

The 300 Area is located approximately two kilometers north of the city of Richland, Washington, in the southeastern section of Hanford site. Because the demand for production reactor fuel has been reduced at Hanford, reactor fuel fabrication operations have become secondary to the research and development activities in the area.

The Pacific Northwest Laboratory (PNL) conducts research in the physical, life, and environmental sciences, environmental surveillance, and advanced methods of nuclear waste management. The Hanford Engineering Development Laboratory (HEDL) is involved in advanced reactor developments.

Each facility within the area has its own treatment system for exhausted air. Radioactive particulates are removed from the exhaust gases by various stages of HEPA filters located near the hood, glovebox or cell where they are generated. Charcoal filters are used where needed to absorb radioiodines. Radioactive atmospheric releases from the 300 Area total less than 1 milicurie.

400 AREA

The 400 area of the Hanford site, the newest area to be developed, is located just north of the 300 area, about 20 kilometers from Richland, Washington, and 7 kilometers from the west bank of the Columbia river.

At the present time there are no sources emitting radioactivity to the atmosphere. However, future sources of radioactive atmospheric emissions include the Fast Flux Test Facility and the High Performance Fuel Laboratory. Future development in support of the Liquid Metal Fast Breeder Reactor program call for a fuels and materials examination facility, maintainance and storage facilities, fuel storage facility and support facilities.

3.1.3 Emission of Radionuclides

Table 3.1-1 summarizes atmospheric emissions from the Hanford site reported in the annual environmental surveillance report for 1977.

Table 3.1-1. Atmospheric emissions of radionuclides, Hanford Site, 1977 (Ho78)

Radionuclide	Emissions (Ci/y)			
	100 Area	200 Area	300 Area	400 Area[a]
Tritium	1.8E+1	-	9.0	1.6E+1
Sodium-21	1.8E-1	-	-	-
Phosphorus-32	-	-	-	-
Argon-41	1.3E+5	-	-	-
Chromium-51	1.7E-2	-	-	-
Maganese-54	1.6E-2	-	-	-
Maganese-56	2.4	-	-	-
Iron-59	1.8E-2	-	-	-
Cobalt-58	2.9E-3	-	-	-
Cobalt-60	2.9E-2	-	[b]1.2E-4	-
Zinc-65	4.5E-4	-	-	-
Arsenic-76	6.6E-1	-	-	-
Krypton-85m	8.3E+2	-	-	-
Krypton-87	2.5E+3	-	-	-
Krypton-Rubidium-88	2.8E+3	-	-	-
Strontium-89	7.9E-3	-	-	-
Strontium-90	1.7E-4	[c]2.1E-1	[c]2.5E-4	-
Strontium-91	5.8E-1	-	-	-
Zirconium-95	3.8E-3	-	-	-
Niobium-95	3.0E-3	-	-	-
Zirconium-Niobium-97	2.1E-3	-	-	-
Molybodenum-Technetium-99	6.2E-1	-	-	-
Ruthenium-103	8.2E-3	-	-	-
Ruthenium-106	1.9E-2	-	-	-
Antimony-122	5.4E-3	-	-	-
Antimony-124	3.3E-3	-	-	-
Antimony-125	1.4E-4	-	-	-
Tellurium-132	5.6E-3	-	-	-
Iodine-129	2.1E-7	-	-	-

See footnotes at the end of table

Table 3.1-1. Atmospheric emissions, of radionuclides
Hanford Site, 1977--continued

Radionuclide	Emissions (Ci/y)			
	100 Area	200 Area	300 Area	400 Area[a]
Iodine-131	5.5E-1	-	4.4E-4	-
Iodine-132	9.6	-	-	-
Iodine-133	4.0	-	-	-
Iodine-135	8.6	-	-	-
Xenon-133	6.8E+2	-	-	-
Xenon-135	3.4E+3	-	-	-
Cesium-137	1.5E-3	-	-	-
Cesium-138	1.3E+4	-	-	-
Barium-140	2.0E-1	-	-	-
Lanthanum-140	3.6E-1	-	-	-
Cerium-141	1.3E-3	-	-	-
Cerium-Praseodymium-144	3.0	-	-	-
Neodymium-147	1.3E-2	-	-	-
Samarium-153	1.7E-3	-	-	-
Europium-154	1.0E-2	-	-	-
Europium-155	6.2E-3	-	-	-
Tungsten-187	6.9E-2	-	-	-
Uranium-238	-	-	5.2E-5	-
Neptunium-239	3.2E-3	-	-	-
Plutonium-238	1.1E-6	-	-	1.2E-5
Plutonium-239	5.7E-5	[d]2.8E-3	[d]3.2E-5	1.1E-6
Plutonium-240	-	-	-	2.2E-6
Americium-241	4.0E-3	-	-	1.4E-6
Plutonium-241	-	-	-	1.4E-6
Plutonium-242	-	-	-	5.3E-4
Curium-244	-	-	8.4E-8	1.1E-8

[a] Estimated from available environmental statements.

[b] Actually reported as mixed activation products. Cobalt-60 was assumed to be the radionuclide for dose calculations.

[c] Actually reported as mixed fission products. Strontium-90 was assumed to be the radionuclide for dose calculations.

[d] Actually reported as gross alpha. Plutonium-239 was assumed to be the radionuclide for dose calculations.

3.1.4 Health Impact Assessment at Hanford Site.

The maximum site boundary dose from 1977 atmospheric emissions at the Hanford site occurred along the northwest boundary. The annual total body dose to an individual at the site boundary was reported to be 0.14 mrem. The principal radionuclides contributing to this dose were cesium-138 and argon-41 from N reactor operations in the 100 Area

Table 3.1-2 estimates individual doses resulting from one year of operations and following 50 years of operations at the site. The estimated maximum individual dose would be received at a location 1.6 kilometers east of the 300 Area and was .06 mrem to the thyroid. The 50-year dose commitment to the maximum exposed individual was .07 mrem to the bone.

Because the population affected by the emissions from each area differs greatly, the 80-kilometer radius population dose was calculated for each area. The annual 50-year dose commitments are reported in table 3.1-3. The highest population dose was estimated to be 4.1 person-rems to the bone from N reactor operations in the 100 Area following 50 years of operations.

Table 3.1-4 estimates the individual lifetime risks and the number of fatal cancers resulting from these doses. The lifetime cancer risk to the maximum individual is estimated to be 2.7 x 10^{-7}. The estimated number of fatal cancers per year of Hanford operation to the population within 80 kilometers is estimated to be about 6.5 x 10^{-4}.

Table 3.1-2. Doses from radioactive emissions, Hanford Site, 1977 (Ho78)

Organ	Maximum individual[a] (mrem)	
	Annual	50-year dose commitment
Lung	.03	.03
Bone	.02	.07
Thyroid	.06	.06
G.I. Tract	.03	.03
Skin	.03	.03
Total body	.03	.03

[a] The maximum individual is located 1.6 km east of 100 Area.

Table 3.1-3. Population doses from radioactive emissions, Hanford Site, 1977 (Ho78)

Organ	Population dose (person-rem)[a]					
	Annual			50-year Commitment		
	100-N Area	200 Area	300 Area	100 Area	200 Area	300 Area
Lung	2.1	.02	.01	2.1	.06	.02
Bone	2.0	.02	.01	2.4	1.5	.15
Thyroid	4.0	.01	.03	4.1	.01	.03
G.I. Tract	1.9	.02	.01	1.9	.02	.01
Total body	2.0	.01	.01	2.0	.16	.02

[a] Population within 80 kilometers: 100-N Area, 236,000; 200 Area, 258,000; 300 Area, 171,000.

Table 3.1-4. Individual lifetime risks and number of fatal cancers from radioactive emissions, Hanford Site, 1977

Organ	Individual lifetime risks Maximum individual[a]	Expected fatal cancers per year of operation[b] (Fatal cancers)
Lung	8.4E-8	8.5E-5
Bone	2.8E-8	8.1E-5
Thyroid	4.2E-9	4.0E-6
G.I. Tract	4.2E-8	3.9E-5
Other soft tissue	1.1E-7	4.4E-4
Total	2.7E-7	6.5E-4

[a] The maximum individual is located 1.6 km east of 300 Area.

[b] The population is within 80 km.

REFERENCES

AEC72 Atomic Energy Commission, 1972, Environmental Statement, WASH-1510, Fast Flux Test Facility, Richland, Washington.

AEC74 Atomic Energy Commission, 1974 Draft Environmental Statement, WASH-1538, Waste Management Operations, Hanford Reservation, Richland, Washington.

ERDA77a Energy Research & Development Administration, 1977 Draft Environmental Impact Statement, ERDA 1556-D, High Flux Neutron Source Facility, Hanford Reservation, Richland, Washington.

ERDA77b Energy Research and Development Administration, 1977 Final Environmental Impact Statement, ERDA-1550, High Performance Fuel Laboratory, Hanford Reservation, Richland, Washington.

Ho78 Houston J. R. and Blumer P. J., 1978, Environmental Surveillance at Hanford for CY-1977, PNL-2614, Battelle Pacific Northwest Laboratories, Richland, Washington

3.2 Savannah River Plant

3.2.1 General Description

Savannah River Plant (SRP) facilities were established in the early 1950's to produce nuclear materials for national defense requirements, principally plutonium and tritium. The site is 150 kilometers from Savannah, Georgia, on a 78,000-hectare tract along the Savannah River in Barnwell, Allendale, and Aiken Counties, South Carolina. The population density varies from 4 to over 154 people per square kilometer in the surrounding area, which is predominantly rural with forested countryside and diversified farming.

Major operating facilities include three nuclear reactors, two chemical separations plants, a fuel and target fabrication plant, a heavy water production plant, and the Savannah River Laboratory. These facilities, which are the principal sources of radioactive airborne effluents, are centrally located on the site, while the administration and other non-nuclear support facilities are located nearer the site periphery.

3.2.2 Process Description

Nuclear Reactors

Three operating nuclear reactors (C, K, and P) produce plutonium and tritium for nuclear weapons. Two additional reactors have been shut down and placed in standby operational status. All five reactors are fueled with uranium and are moderated with heavy water. During 1977, the three operating reactors released large amounts of tritium, argon-41, krypton and xenon gases, and carbon-14 to the atmosphere. At each reactor the radioactive atmospheric effluents are treated by HEPA and charcoal filtration before being released from a 61-meter stack. This filtration system removes only radioactive particles and iodines and has no effect on the noble gases and tritium (which is released as tritiated water vapor).

Chemical Separations Facilities

The chemical separations facilities consist of two separate areas, the F- and H-areas, which process irradiated fuel and uranium targets. In the F-area, plutonium-239 is recovered using the Purex solvent extraction process. The F-area also contains the main analytical laboratory, the plutonium metallurgical laboratory, and the Plutonium Fuel Form Facility.

The H-area canyon building is similar to the F-area separations building. Special radionuclides and uranium-235 are recovered using a solvent extraction process like the Purex process, but with modifications depending on the nature of the fuels and the radionuclide to be recovered. The H-area also contains the Tritium Processing Building, the Receiving Basin for Offsite Fuels and the Resin Regeneration Facility.

The major releases from the separations area include tritium, krypton-85, carbon-14, xenon gases, and other fission products. The primary sources of radioactivity in the effluent are from the main process canyon air, the vessel vent system off-gases, and the plutonium powder-handling cabinets. The effluent is passed through a deep-bed sand filter and is then released. Ventilation air from the separations area may undergo several stages of filtration for removal of particulate radioactivity prior to release.

The F-area main processing, or canyon building, is served by a 61-meter stack, which releases the combined filtered effluents from several sections of the building. Similarly, the H-area processing canyon building is serviced by a 61-meter stack and the main process canyon air and the process vessel vent off-gases are routed through a deep-bed sand filter. Both stacks are continuously monitored.

Other sources of releases of radioactivity to the atmosphere from the H-area include: (1) three tritium facilities which are serviced by three 61-meter stacks and one 23-meter stack; (2) the receiving basin for off-site Fuels and Resin Regeneration Facility which releases fission product gases from shipping casks and fuel cutting and cleaning operations--these effluents are released from a short stack, 16 meters above grade; and (3) the ventilation air from the metallurgical building and analytical laboratory which is filtered through HEPA filters and released from 23-meter stacks.

Fuel and Target Fabrication Facilities

The Fuel and Target Fabrication Facilities make enriched uranium-aluminum alloy fuel and canned depleted uranium metal targets for the Savannah River Plant reactors. During 1977, the three major buildings in this area released small amounts of natural uranium and unidentified alpha activity. Off-gas exhausts from the various operations and laboratories are filtered through HEPA filters, monitored, and then released from 15-meter, 23-meter, and several 30-meter stacks.

Heavy Water Production Facility

The heavy water production and recovery facilities emit tritium into the atmosphere. Evaporated tritiated heavy water from the rework unit pump room and drum handling building is vented to a 21-meter stack. Radioactivitiy releases are monitored continuously. Monthly estimates of atmospheric releases from the drum cleaning facility and the analytical laboratory are based on periodic grab samples.

Savannah River Laboratory

The Savannah River Laboratory, which is responsible for research and development activities at SRP, releases small amounts of tritium, cobalt-60, and iodine-131. HEPA and charcoal filtration treat the radioactive effluents which are then monitored and released from eight stacks, 6 to 23 meters high.

3.2.3. Emission of Radionuclides

Emissions from the Savannah River Plant in 1977 are given in Table 3.2-1.

Table 3.2-1. Atmospheric emissions from the Savannah River Plant, 1977 (DOE77)

Facility and Radionuclide	Emissions (Ci/y)
C Reactor Plant	
Argon-41	3.1E+4
Carbon-14	1.2E+1
Tritium	8.3E+4
Iodine-131	4.9E-3
Krypton-85m	1.8E+2
Krypton-87	2.2E+2
Krypton-88	3.3E+2
Unidentified Alpha	2.6E-6
Unidentified Beta and Gamma	1.8E-4
Xenon-133	1.2E+3
Xenon-135	7.8E+2
K Reactor Plant	
Argon-41	1.6E+4
Carbon-14	1.3E+1
Tritium	4.2E+4
Iodine-131	3.3E-4
Krypton-85m	4.9E+1
Krypton-87	6.4E+1
Krypton-88	7.2E+1
Unidentified Alpha	1.9E-6
Unidentified Beta	1.8E-4
Xenon-133	2.4E+2
Xenon-135	1.1E+2
P Reactor Plant	
Argon-41	1.8E+4
Carbon-14	1.2E+1
Tritium	3.9E+4
Iodine-131	1.2E-3
Krypton-85m	6.1E+2
Krypton-87	3.1E+2
Krypton-88	2.7E+2
Unidentified Alpha	2.4E-6

Table 3.2-1. Atmospheric emissions from the Savannah River Plant, 1977 (DOE77)--continued

Facility and Radionuclide	Emissions (Ci/y)
P Reactor Plant--continued	
Unidentified Beta and Gamma	1.1E-4
Xenon-133	9.0E+2
Xenon-135	6.5E+2
200 Separations Area	
Tritium	1.9E+5
Iodine-129	1.4E-1
Krypton-85	4.4E+5
Xenon-131m	1.2E+1
Xenon-133	1.3E-2
F-Area Separations Plant	
Americium-241	1.5E-4
Carbon-14	1.3E+1
Cerium-141	2.1E-4
Cerium-144	4.8E-3
Curium-244	1.5E-4
Cesium-134	2.0E-6
Cesium-137	5.6E-4
Iodine-131	3.8E-2
Niobium-95	3.3E-2
Plutonium-238	2.8E-4
Plutonium-239	1.0E-4
Ruthenium-103	1.5E-3
Ruthenium-106	1.0E-2
Strontium-89 and Strontium-90	2.0E-3
Uranium-238	1.2E-3
Unidentified Beta and Gamma	1.6E-5
Ziconium-95	7.6E-3

Table 3.2-1. Atmospheric emissions from the Savannah River Plant, 1977 (DOE77)--continued

Facility and Radionuclide	Emissions (Ci/y)
H-Area Separations Plant	
Americium-241	1.9E-4
Carbon-14	1.3E+1
Cerium-141	1.4E-4
Cerium-144	1.0E-2
Curium-244	1.9E-4
Cesium-134	4.3E-4
Cesium-137	1.3E-3
Iodine-131	1.2E-2
Niobium-95	4.4E-3
Plutonium-238	4.7E-3
Ruthenium-103	2.3E-2
Ruthenium-106	1.3E-1
Strontium-89 and Strontium-90	2.2E-3
Uranium-238	3.2E-4
Unidentified Beta & Gamma	4.0E-6
Zirconium-95	6.8E-3
Fuel and Target Fabrication Plant	
Uranium-238	1.8E-5
Unidentifed Alpha	1.7E-5
Heavy Water Production Plant	
Tritium	2.9E+3
Savannah River Laboratory	
Cobalt-60	3.8E-4
Tritium	4.0
Iodine-131	5.0E-3
Unidentified Alpha	1.0E-6
Unidentified Beta & Gamma	7.0E-6

3.2.4 Dose Assessment of the Savannah River Plant

The 70-year dose commitment from a one-year release of radioactive materials from the Savannah River Plant, based on 1977 radioactive atmospheric releases, was calculated for the maximum exposed individual at the site boundary and for the population within 80 kilometers. The major contributors to the maximum exposed individual whole-body dose at the perimeter were tritium, 78 percent; argon-41, 15 percent; carbon-14, 5 percent; and from krypton, xenon, iodine-129, iodine-131, and other radionuclides, 2 percent. The population dose commitment calculated for the population within an 80-kilometer radius was 114 person-rem, with tritium accounting for 86 percent; argon-41, 7 percent; and carbon-14, 6 percent.

Table 3.2-2 summarizes the contribution of each nuclide to the maximum site boundary dose and the 80-kilometer population dose. The doses shown are the 70 year lifetime dose commitment based on 1977 atmospheric emissions.

Table 3.2-3 estimates the individual lifetime risks and the number of fatal cancers resulting from these doses. The lifetime cancer risk to the maximum individual is estimated to be 1.5E-5. The estimated number of fatal cancers per year of SRP operation to the population within 80 kilometers is estimated to be about 2.2E-2.

Table 3.2-2. Site boundary and population dose commitments[a] from atmospheric emissions at Savannah River Plant, 1977 (DP78)

Radionuclide	Maximum individual (mrem)	Population (person-rem)
Tritium	8.5E-1	9.8E+1
Carbon-14	6.0E-2	6.9
Argon-41	1.8E-1	8.3
Krypton-85m	4.9E-4	3.2E-2
Krypton-85	2.2E-3	2.7E-1
Krypton-87	1.7E-3	6.4E-2
Krypton-88	4.3E-3	2.4E-2
Xenon-131m	1.0E-5	8.0E-4
Xenon-133	8.6E-4	8.2E-2
Xenon-135	1.5E-3	1.1E-1
Iodine-129	1.3E-5	1.0E-1
Iodine-131	1.0E-5	1.2E-3
Cobalt-60	1.0E-5	1.0E-4
Strontium-89 and Strontium-90	1.0E-5	1.0E-4
Zirconium-95	1.0E-5	1.0E-4
Niobium-95	1.0E-5	1.0E-4
Ruthenium-103	1.0E-5	1.0E-4
Ruthenium-106	1.0E-5	1.0E-4
Cesium-134	1.0E-5	1.0E-4
Cesium-137	1.0E-5	1.0E-4
Cerium-141	1.0E-5	1.0E-4
Cerium-144	1.0E-5	1.0E-4
Uranium	1.0E-5	1.0E-4
Plutonium-238	5.3E-4	3.6E-2
Plutonium-239	1.0E-5	8.0E-4
Americium-241	1.0E-5	8.0E-4
Curium-244	1.0E-5	8.0E-4
	Totals 1.1	1.1E+2

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[a] A 70-year dose commitment based on 1977 atmospheric emissions of radioactive materials. The maximum exposed individual is at the site boundary; the population is within 80 kilometers of SRP.

Table 3.2-3. Individual lifetime risks and number of fatal cancers[a] from radioactive emissions, Savannah River Plant, 1977

Source	Individual lifetime risks		Expected fatal cancers per year of operation (Fatal cancers)
	Maximum individual[b]	Average individual Region	
SRP	1.5E-5	3.3E-6	2.2E-2

[a] To the population within 80 kilometers.
[b] The maximum individual is at the site boundary.

REFERENCES

DOE77 Department of Energy, 1977, Effluent Information System Report No. 51, Release Point Analysis Report for 1977, EIS-51, Washington, D.C.

ERDA77 Energy Research & Development Administration, 1977, Final Environmental Impact Statement, Waste Management Operations, Savannah River Plant, Aiken, South Carolina, ERDA-1537, UC-2-11-70,

DP78 E.I. du Pont de Nemours & Company, 1978 Environmental Monitoring in the Vicinity of the Savannah River Plant, Annual Report for 1977, DPSU 78-30-1.

3.3 Idaho National Engineering Laboratory

3.3.1 General Description

The Idaho National Engineering Laboratory (INEL) was established in 1949 for operating and testing nuclear reactors and critical facilities. Today, a broad scope of engineering activities are conducted there. Seventeen reactors and critical facilities support the following programs: naval propulsion, fast breeder development, lightwater reactor safety testing, organic moderator and coolant development, materials testing, portable military power space development, decontamination and decommissioning, spent fuel reprocessing, waste management and miscellaneous research.

The INEL site, occupying more than 230 thousand hectares on the Upper Snake River Plain in southeastern Idaho, is 48 kilometers west of Idaho Falls, Idaho. The surrounding area is a grazing range for livestock with some irrigation farming to the north and northeast. The total population within 80 kilometers of INEL is approximately 94,000.

3.3.2 Process Description

There are currently four primary contractors which operate the facilities for the Department of Energy: EG & G Idaho, Allied Chemical Corporation, Westinghouse Electric Corporation, and Argonne National Laboratory.

EG & G Idaho Facilities

EG & G Idaho operates the Test Reactor Area, the Test Area North, the Loss-of-Fluid-Test Facility, the Power Burst Facility, the Auxilary Reactor Area, the INEL Radioactive Waste Management Complex and the Central Facilities Area. The major programs conducted by EG & G Idaho include test irradiation and light-water-cooled reactor safety testing and research.

Test Reactor Area

The Test Reactor Area, in the south central part of INEL, provides facilities for studying the performance of reactor materials and equipment components under high neutron flux conditions. The reactor area contains three reactors, the Materials Testing Reactor (MTR), the Engineering Test Reactor (ETR) and the Advanced Test

Reactor (ATR). Also, there are three low-power reactors located in this area, the Engineering Test Reactor Critical facility, the Advanced Test Reactor Critical facility and the Advanced Reactivity Measurement Facility.

The MTR is in a standby status. Exhaust air from these laboratories is treated by caustic scrubbing and HEPA filtration. Exhausted air is released from a 26-meter stack after being monitored. No detectable release of radioactivity has been observed from the MTR since its shutdown.

Ventilation air from the ATR and ETR is discharged from 76-meter stacks at each building. No waste treatment system is employed at either of these facilities. The stacks are monitored continuously for both gaseous and particulate activity. Large amounts of argon-41 and radioactive isotopes of krypton and xenon are released from these facilities.

Test Area North (TAN)

The Technical Support Facilities (TSF) occupy 28 hectares and consist of about 20 buildings. The TSF supports water reactor safety programs, particularly the Loss-of-Fluid Test activities. Primary sources of radioactive airborne releases from the TSF area include hot shops area, radioactive material storage pool, warm shop, cleaning rooms and various laboratories.

Each area that has a potential for generating radioactive airborne waste has its own effluent control system which includes roughing and HEPA filters. No provisions have been made for removal of radioiodines or for monitoring radioactive gaseous wastes such as isotopes of krypton and xenon and argon-41. The effluents are monitored for alpha and beta activity. Small amounts of fission products constitute typical releases from the TSF area.

The Low Power Test (LPT) and Experimental Beryllium Oxide Reactor (EBOR) facilities are also located in the TAN area. There is little potential for radioactive airborne releases from these facilities because of minimal reactor operations at the LPT and only non-nuclear testing conducted at the EBOR. However the exhaust systems at each facility are equipped with prefilters and absolute HEPA filters.

Loss-of-Fluid Test (LOFT) Facility

The LOFT facility, located in the TAN complex, is the only reactor in operation in this area. Radioactive airborne effluents from reactor operations undergo several stages of filtration which include roughing and HEPA filters. Charcoal absorbers are used to remove halogens. Airborne effluents are exhausted from a 46-meter stack which is monitored for gases and particulates. No radioactive airborne effluents were reported in 1977.

Power Burst Facility

The Power Burst Facility is a high-performance, water-cooled, uranium oxide fueled reactor used in support of the light-water reactor safety testing program. Airborne effluents are vented through roughing and HEPA filters. Silver zeolite absorbers are used for iodine removal. Ventilation air is then released from a 24-meter stack equipped with a gas monitor and particulate sampler. Small quantities of activation and fission products are released.

Auxiliary Reactor Area (ARA)

The ARA area is located to the southeast of the PBF in the south-central section of the INEL site. This was the location of the Army reactors which have been shutdown. Only the ARA-1 hot cells remain as a source of radioactive airborne releases. The exhaust system serving the ARA-1 hot cells is equipped with roughing and HEPA filters. One cell utilizes charcoal filters for iodine removal. The exhaust stack is continuously monitored for iodine, particulate, alpha, and gaseous activity. The annual release of mixed activation and fission products is very small.

Radioactive Waste Management Complex

Located in the southwest corner of the INEL site, this complex disposes solid low-level radioactive waste materials generated by laboratory operations. There is a subsurface disposal area used for burial of nontransuranic solid wastes. Transuranic wastes in containers having less than 10 nCi/g of activity are stacked on an asphalt pad and covered with earth in the separate Transuranic Disposal Area. A third area, the Transuranic Storage Area, provides interim storage for containers having greater than 10 nCi/g of transuranic acitivity. Operations in this area are monitored to detect any release of hazardous materials. There are no routine releases reported.

Central Facilities Area

This area, located in the south-central portion of the site, provides support services for outlying operational areas. The only potential source for airborne radioactivity release is from the laundry facility. The off-gas vent from the dryer is equipped with screens and filters to prevent the spread of contaminated lint. The area is surveyed frequently.

Allied Chemical Corporation

The Allied Chemical Corporation operates the Idaho Chemical Processing Plant (ICPP), the Waste Calcining Facility (WCF), the Fuel Storage Basin Facility and the Tank Farm Facility. Of these facilities, the ICPP and the WCF are the primary sources of airborne radioactivity.

At the ICPP, enriched uranium is recovered from spent nuclear fuel by a solvent extraction process. The major sources of radioactive airborne effluent are the process dissolvers, process vessels and areas, analytical facilities, sample stations, waste solvent burners and process ventilation air. Process off-gas streams from the dissolvers are routed through a reflux condenser, entrainment separator, demister, superheater and HEPA filters before being released from the stack. Two of the three dissolver off-gas streams can be sent to the Rare Gas Plant where radioactive xenon and krypton can be recovered by a cryogenic distillation process. The remaining off-gases are collected, then routed through a demister, superheater and HEPA filter prior to discharge. Process sampling stations exhaust to the stack through fiber glass filters. The waste solvent burner exhaust and process ventilation air flow directly to the stack without any treatment.

The Waste Calcining Facility (WCF) solidifies liquid waste generated by the ICPP. This process results in significant amounts of airborne activity. The WCF employs an extensive cleanup system to remove particulates and volatile fission products. The cleanup system includes a cyclone, scrubbing system, silica gel absorber beds and HEPA filters. Radiation detectors monitor the process off-gases which are then exhausted to the ICPP stack.

The 76-meter ICPP stack is monitored for gross activity, and radiochemical analyses are conducted on daily filter samples. From these measurements, the total activity in the emissions from the stack are estimated. In 1977, krypton-85 and tritium were the principal nuclides contributing to the total activity released.

Westinghouse Electric Corporation

Westinghouse Electric Corporation operates the Naval Reactor Facility (NRF), which consists of three operating naval reactor prototypes (S1W, A1W and S5G) and the Expended Core Facility (ECF). The NRF area is located in the south-central area just north of the Test Reactor Area.

The AIW reactor plant radioactive waste system building, the S5G reactor building and the ECF are equipped with HEPA filters in the exhaust systems. The principal nuclides released from the NRF complex were several curies of krypton-85 and xenon-133 which originated from the ECF; very small quantities of mixed fission and activation products were released from the reactor buildings.

Argonne National Laboratory-West (ANL-W)

The Argonne National Laboratory-West facilities are located in the southeastern portion of the INEL site. There are five major complexes which comprise ANL-W; the Experimental Breeder Reactor No. 2, the Transient Reactor Test Facility, the Zero Power Plutonium Reactor, the Hot Fuel Examination Facility, and the Laboratory and Office and support complex.

Experimental Breeder Reactor No. 2 (EBR-II)

The EBR-II is an experimental liquid-metal cooled fast breeder reactor used to irradiate samples of reactor fuels and structural material for the LMFBR development program. Associated with the EBR-II are the sodium component cleanup facility and a fuel assembly and storage building.

The primary tank of the reactor is filled with an argon cover gas. Cover gas that leaks into the containment building is withdrawn along with the ventilation air through HEPA filters. The argon covergas can be purged directly to the atmosphere, bypassing the containment building and the HEPA Filters. The containment building discharge may be initiated by the gamma monitoring system.

Radioactive airborne effluents from the EBR-II consist primarily of fission product noble gases, which consist of isotopes of xenon, krypton and argon-41.

The sodium component cleanup facility releases small amounts of radioactivity. Airborne effluents are treated by HEPA filtration before release from a local stack. Monthly filter samples are used to determine gross alpha and beta activity released.

The Fuel Assembly and Storage Building (FASB) airborne effluents are treated by HEPA filtration. Airborne effluents resulting from fuel assembly operations are monitored by monthly filter samples for gross alpha, beta and gamma activity.

Transient Reactor Test Facility (TREAT)

The TREAT facility reactor produces short extreme pulses of nuclear energy for melt-down studies of prototype and experimental fuel elements. Airborne effluents from the reactor operations are exhausted through HEPA filters and an 18-meter stack. The exhaust air is continuously monitored and cold trap samples are obtained periodically. The TREAT radioactive gaseous effluent consists of isotopes of krypton and xenon and argon-41.

Zero Power Plutonium Reactor (ZPPR)

The ZPPR experiments provide information about reactor physics in support of the plutonium fuel fast breeder reactor program. During reactor operations the cooling air is recirculated through HEPA filters. When the ZPPR is not in operation, the cooling air is exhausted through HEPA filters and an 18-meter stack. The cooling air is continuously monitored both upstream and downstream of the HEPA filters. The radioactive atmospheric effluents from the ZPPR are composed primarily of noble gas fission product isotopes of krypton and xenon.

Hot Fuels Examination Facility (HFEF)

The Hot Fuels Examination Facility consists of two separate buildings. The HFEF-South is used mainly as an irradiation subassembly-disassembly, inspection and assembly point and the HFEF-North, for diagnostics and inspections.

The HFEF-South contains two cells used for processing operations. The facility is used for examination of materials and fuels irradiated in the EBR-II and TREAT. Ventilation air from this facility is treated by HEPA filters and exhausted from the 61-meter EBR-II stack. There are two work areas used for cleanup that have separate exhaust and multiple HEPA filtration systems.

The HFEF-North is used for interim and final examination of fast reactor fuel and structural specimens. The main cell has an argon atmosphere. The ventilation system passes all radioactive airborne effluents through HEPA filters. The exhaust is continuously monitored for alpha and beta activity and selectively for cesium-137, iodine-131, xenon-133 and krypton-85. Only small releases were detected in 1977.

Laboratory and Office Complex

Potential sources of radioactive airborne emissions are from radiochemical hoods in the several laboratories of the Laboratory and Office Complex. The sodium chemistry laboratory is the only building in this complex using its own short stack. Exhaust air from all the other laboratories undergoes HEPA filtration before release from the main EBR-II stack.

3.3.3 Emissions of Radionuclides

Table 3.3.1 summarizes the radioactive airborne emissions from the INEL facilities in 1977.

Table 3.3-1. Atmospheric emissions of radionuclides from the Idaho National Engineering Laboratory Site, 1977

Source	Radionuclide	Emissions (Ci/y)
EG & G IDAHO		
Test Reactor Area	Argon-41	3.2E+3
	Barium-139	4.9E+2
	Cesium-138	7.3E+1
	Krypton-85m	1.1E+3
	Krypton-87	4.8E+3
	Krypton-88	3.4E+3
	Rubidium-88	4.3E+1
	Xenon-133	3.1E+2
	Xenon-135	9.6E+1
	Xenon-135m	3.6E+3
	Xenon-138	1.1E+4
Test Area North	Cerium-144	1.0E-7
	Cobalt-58	5.1E-6
	Cobalt-60	2.5E-5
	Cesium-134	6.2E-7
	Cesium-137	9.7E-5
	Iodine-131	8.9E-7
	Lanthanum-140	6.7E-9
	Manganese-54	8.7E-6
	Ruthenium-103	5.9E-7
	Strontium-90	5.7E-7
	Unidentified Alpha	5.9E-6
	Unidentiffied Beta & Gamma	7.5E-5
Power Burst Facility	Silver-110m	6.3E-7
	Barium-140	9.8E-7
	Cerium-141	9.3E-7
	Cerium-144	1.3E-6
	Cobalt-58	1.5E-7
	Cobalt-60	4.7E-6
	Chromium-51	2.4E-6

Table 3.3-1. Atmospheric emissions of radionuclides from the Idaho National Engineering Laboratory Site, 1977--continued

Source	Radionuclide	Emissions (Ci/y)
Power Burst Facility--continued		
	Cesium-134	1.7E-6
	Cesium-137	3.5E-5
	Iron-59	1.1E-6
	Mercury-203	1.7E-6
	Iodine-131	9.7E-6
	Lanthanum-140	5.1E-6
	Manganese-54	3.0E-6
	Niobium-95	1.1E-6
	Antimony-125	3.5E-7
	Strontium-90	2.1E-6
	Unidentified Alpha	2.2E-7
	Unidentified Beta & Gamma	2.5E-5
	Tungsten-187	9.4E-7
	Zirconium-95	2.4E-7
Auxilary Reactor Area	Cerium-141	2.0E-9
	Cerium-144	1.6E-8
	Cobalt-58	1.8E-8
	Cobalt-60	1.9E-7
	Cesium-134	1.0E-8
	Cesium-137	3.2E-7
	Europium-152	2.8E-7
	Iodine-131	1.3E-6
	Manganese-54	6.1E-8
	Niobium-95	3.1E-8
	Strontium-90	1.6E-7
	Unidentified Alpha	6.1E-9
	Unidentified Beta & Gamma	4.2E-7

Table 3.3-1. Atmospheric emissions of radionuclides from the Idaho National Engineering Laboratory Site, 1977--continued

Source	Radionuclide	Emissions (Ci/y)
Allied Chemical Corporation		
Idaho Chemical Processing Plant (ICPP) and the Waste Calcining Facility (WCF)	Cerium-144	1.02E-1
	Cobalt-60	3.00E-5
	Cesium-134	6.30E-3
	Cesium-137	2.03E-1
	Europium-154	1.51E-4
	Tritium	3.08E+3
	Krypton-85	1.08E+5
	Manganese-54	3.73E-5
	Niobium-95	2.96E-3
	Praseodymium-144	1.02E-1
	Plutonium-238	9.93E-4
	Plutonium-239	3.57E-4
	Plutonium-240	6.30E-5
	Rhodium-106	1.39E-1
	Rubidium-106	1.39E-1
	Antimony-125	3.44E-2
	Strontium-90	6.16E-2
	Thorium-232	2.64E-5
	Yttrium-90	6.16E-2
	Zirconium-Niobium-95	2.00E-3
	Zirconium-95	1.71E-3
Westinghouse Electric Corp.--Naval Reactor Facility		
S1W Reactor Plant	Argon-41	3.9E-4
	Cobalt-60	2.6E-6
	Cesium-137	2.4E-6
	Iodine-131	2.3E-7
	Iodine-132	8.5E-8
	Iodine-133	3.9E-7
	Iodine-134	7.1E-8
	Iodine-135	2.5E-7
	Krypton-85	7.5E-5
	Krypton-88	2.5E-5

Table 3.3-1. Atmospheric emissions of radionuclides from the Idaho Engineering Laboratory Site, 1977--continued

Source	Radionuclide	Emissions (Ci/y)
Westinghouse Electric Corp.--Naval Reactor Facility--continued		
S1W Reactor Plant--continued		
	Unidentified Beta & Gamma	5.8E-6
	Xenon-133	1.7E-3
	Xenon-138	4.1E-6
A1W Reactor Plant and Radioactive Waste System Building		
	Argon-41	1.3E-3
	Cobalt-60	1.8E-5
	Iodine-131	4.5E-7
	Iodine-132	1.9E-7
	Iodine-133	8.9E-7
	Iodine-134	1.7E-7
	Iodine-135	5.7E-7
	Krypton-85	1.2E-4
	Krypton-88	5.7E-5
	Unidentified Beta & Gamma	3.3E-6
	Xenon-133	3.4E-3
	Xenon-138	9.7E-6
S5G Reactor Plant		
	Argon-41	1.2E-3
	Cobalt-60	5.4E-7
	Iodine-131	2.0E-7
	Iodine-132	7.6E-8
	Iodine-133	3.6E-7
	Iodine-134	6.9E-8
	Iodine-135	2.3E-7
	Krypton-85	3.0E-5
	Krypton-88	2.3E-5
	Xenon-133	1.5E-3
	Xenon-138	3.9E-6

Table 3.3-1. Atmospheric emissions of radionuclides from the Idaho Engineering Laboratory Site, 1977--continued

Source	Radionuclide	Emissions (Ci/y)
Expended Core Facility		
	Cobalt-60	2.4E-6
	Tritium	4.1E-3
	Iodine-131	1.7E-3
	Krypton-85	1.8
	Xenon-133	8.2
Argonne National Laboratory-West		
EBR-II; HFEF-S; Laboratory & Office	Argon-41	7.4
	Bromine-82	1.2E-1
	Tritium-3	6.7E-1
	Krypton-85	5.0
	Krypton-85m	2.1
	Krypton-87	1.2
	Krypton-88	1.7
	Rubidium-88	8.4E-3
	Xenon-133	3.4E+2
	Xenon-135	1.3E+2
	Xenon-135m	2.1E-3
	Xenon-138	9.4E-3
Sodium Components Cleanup Facility	Cesium-137	1.3E-7
	Unidentified Alpha	1.8E-7
	Unidentified Beta & Gamma	1.1E-6
Fuel Assembly & Storage Area	Unidentified Alpha	2.1E-7
	Unidentified Beta & Gamma	1.5E-6

Table 3.3-1. Atmospheric emissions of radionuclides from the Idaho Engineering Laboratory Site, 1977--continued

Source	Radionuclide	Emissions (Ci/y)
Transient Reactor Test	Argon-41	1.4E+2
	Krypton-88	2.2
	Xenon-133	2.7E-1
	Xenon-135	5.4E-1
Zero Power Plutonium Reactor Facility	Krypton-85M	5.8E-1
	Krypton-87	1.1
	Krypton-88	1.2
	Xenon-133	1.1E-2
	Xenon-135	4.3E-1
Hot Fuel Examination Facility-North	Cesium-137	2.1E-7
	Unidentified Alpha	2.5E-7
	Unidentified Beta & Gamma	1.5E-5
Radiological & Environmental Sciences Laboratory	Cesium-134	1.0E-1

3.3.4 Health Impact Assessment of the INEL Site

The maximum total body dose to a hypothetical individual at the site boundary from radioactive emissions was reported to be 0.10 mrem for 1977. This calculated dose is the 50 year dose commitment for chronic exposure during 1977. The maximum concentrations leading to this dose occurred along the southern site boundary and about 94 percent of this dose was found to be due to noble gases and particulates of isotopes having half-lives of less than 10 hours.

The maximum potential dose to an individual of population group was reported to be 0.074 mrem/yr. Terreton-Hamer was the population group reported to have the greatest potential dose from site operations. The estimated life time risk of fatal cancer for a resident of this town is about 1.0E-6.

The estimated dose of 0.90 person-rem/y to the population within 80 kilometers of INEL resulted from radioactive emissions from operations at the Test Reactor Area and the Idaho Chemical Processing Plant which emit more than 98 percent of the total radioactive emissions from the Idaho site. The number of cancers per year of operations is estimated to be 1.8E-4 for the population living in the region of INEL.

Table 3.3-2. Radiation doses[a] due to radioactive emissions from Idaho National Engineering Laboratory (DOE77)

Source	Maximum individual Site boundary (mrem)	Maximum individual Resident of Terreton-Hamer (mrem)	Population[b] (person-rem)
INEL	0.10	0.074	0.90

[a] Fifty-year dose commitment.
[b] The population is within 80 km.

Table 3.3-3. Individual lifetime risks and number of fatal cancers[a] from INEL operations, 1977

Source	Individual lifetime risks			Expected fatal cancers per year of operation (Fatal cancers)
	Site boundary	Resident of Terreton-Hamer	Region	
INEL	1.4E-6	1.0E-6	1.3E-7	1.8E-4

[a] To population within 80 kilometers

REFERENCES

DOE77a Department of Energy, 1977, Effluent Information System Report No 02, Narrative Summary Data Base Master List (EIS 02).

DOE77b Department of Energy, 1977, Effluent Information System Report No 51. Release Point Analysis Report for Calendar year 1977 (EIS 51).

DOE78 Department of Energy, 1978, 1977 Environmental Monitoring Program Report for Idaho National Engineering Laboratory Site.

ERDA77 Energy Research and Development Administration, 1977, Final Environmental Impact Statement, ERDA-1536, Waste Management Operations, Idaho National Engineering laboratory, Idaho.

3.4 Los Alamos Scientific Laboratory

3.4.1 General Description

The Los Alamos Scientific Laboratory (LASL) was established in 1943 for the purpose of nuclear weapons research and development. Today, in addition to weapons research, LASL conducts national security programs including laser fusion, nuclear materials research, and laser isotope separation. Research is also conducted in areas of power reactors, magnetic fusion, radiobiology and medicine, astrophysics, earth sciences, energy resources, lasers and the environment.

LASL is located in Los Alamos County, New Mexico, in the north-central area of the State, near the towns of White Rock and Los Alamos, about 100 kilometers north-northeast of Albuquerque and 40 kilometers northwest of Sante Fe. LASL facilities occupy 11,100 hectares in a relatively undeveloped area on top of the Pajarito Plateau. About 98,000 people reside within 80 kilometers of the laboratory facilities.

3.4.2 Process Description

A wide variety of facilities support the research and devopment programs at Los Alamos Scientific Laboratory. These facilities are located within 30 Technical Areas throughout the LASL site. The major facilities at LASL include an 800 MeV proton accelerator (TA-53), a Tandem Van de Graaff accelerator (TA-3), the laser and magnetic fusion laboratories, and a 10 MW research reactor (TA-2).

Airborne radioactive effluents are discharged from a number of facilities. Those facilities which conduct operations that could have significant releases use appropriate treatment methods to control releases of radioactivity. Hot cells, laboratory hoods and glove boxes are equipped with filtration systems to remove particulate radioactivity. HEPA filters are the principal type of filter used where plutonium and uranium handling operations are conducted. Charcoal filters, bag filters and cyclone separators are used where applicable. Where feasible, tritium releases are reduced from effluents by catylysts, microsieves, and adsorbers. Short-lived activation gas releases are delayed to allow reduction by radioactive decay.

3.4.3 Emissions of Radionuclides

Radioactive airborne emissions are released in stack exhausts from eleven (11) principal Technical Areas. The quantities shown in table 3.4-1 may have significant year-to-year variations depending on the research being conducted.

3.4.4 Health Impact Assessment of Los Alamos Scientific Laboratory

The maximum whole body dose to a hypothetical individual located at the site boundary of LASL was calculated to be 69 mrem for 1977 (table 3.4-2). This dose was primarily due to activation products released in airborne effluents from the proton accelerator. This dose would correspond to a lifetime risk of fatal cancer of 9.7E-4 (table 3.4-3).

The estimated whole body dose to the nearest individual in a population group was calculated to be 19 mrem for 1977. This dose occurred to the north of Technical Area 53 and was due to the activation gases in the effluent from the proton accelerator facility. The lifetime risk to an individual receiving this dose is estimated to be 2.7E-4.

The whole body population dose to the residents within an 80-kilometer radius of LASL was calculated to be 11.1 person-rem for 1977. The activation products argon-41, carbon-11, nitrogen-13 and oxygen-15 were the principal contributors to the dose. The number of fatal cancers per year of LASL operation is estimated to be 2.2E-3 to the population within 80 kilometers.

Table 3.4-1. Atmospheric emissions of radionuclides, Los Alamos Scientific Laboratory, 1977 (DOE77b)

Source	Radionuclide	Emissions (Ci/y)
Technical Area 2 (TA-2) Omega Site	Argon-41	3.2E+2
Technical Area 3 (TA-3)		
Chemical Metallurgical Research Building	Iodine-131	8.8E-5
	Mixed Fission Products	4.8E-4
	Plutonium-238 and Plutonium-239	2.6E-5
	Plutonium-239	6.4E-6
	Uranium-235 and Uranium-238	3.6E-5
Press Building	Uranium-235	2.4E-6
Sigma Building	Thorium-234	5.2E-3
	Uranium-235	3.9E-6
	Uranium-235and Uranium-238	1.8E-6
	Uranium-238	2.8E-4
Technical Shops Addition	Uranium-235 and Uranium-238	3.3E-6
Rolling Mill Building	Uranium-235 and Uranium-238	6.6E-6
Van de Graaff Facility	Tritium	4.0E+2
Technical Area 9 (TA-9) Anchor Test Site	Tritium	2.6

See footnotes at end of table.

Table 3.4-1. Atmospheric emissions of radionuclides, Los Alamos Scientific Laboratory, 1977 (DOE77b)--continued

Source	Radionuclide	Emissions (Ci/y)
Technical Area 21 (TA-21) DP Site	Tritium	1.3E+2
	Mixed Fission Products	3.3E-6
	Plutonium-238	2.7E-7
	Plutonium-238 and Plutonium-239	3.9E-6
	Plutonium-239	5.8E-6
	Uranium-235	3.2E-4
Technical Area 33 (TA-33) HP Site	Tritium	[a]3.7E+4
Technical Area-35 (TA-35) Ten Site	Tritium	7.9E+2
	Plutonium-239	8.2E-7
Technical Area-43 (TA-43) Health Research Laboratory	Phosphorus-32	3.0E-4
	Plutonium-239	4.7E-6
Technical Area-46 (TA-46) WA Site	Plutonium-239	3.0E-9
	Uranium-238	4.0E-9
Technical Area-48 (TA-48) Radiochemistry Site	Mixed Fission Products	2.2E-3
	Plutonium-239	8.4E-6
	Uranium-235	5.5E-5
Technical Area-50 (TA-50) Liquid Waste Treatment Plant	Mixed fission Products	8.6E-5
	Plutonium-239	7.0E-5

See footnotes at end of table.

Table 3.4-1. Atmospheric emissions of radionuclides, Los Alamos Scientific Laboratory, 1977 (DOE77b)--continued

Source	Radionuclide	Emissions (Ci/y)
Technical Area-50 (TA-50) Liquid Waste Treatment Plant	Mixed fission Products	8.6E-5
	Plutonium-239	7.0E-5
Technical Area-53 (TA-53) Meson Physics Facility	Argon-41	4.8E+2
	Beryllium-7	2.8E-7
	Carbon-11	1.4E+4
	Tritium	2.9E+2
	Mixed Activation Products	5.0E-9
	Nitrogen-13	1.4E+3
	Oxygen-15	3.2E+4

[a] Includes 30800 Ci accidental release on October 6, 1977.

Table 3.4-2. Annual radiation doses due to radioactive emissions from Los Alamos Scientific Laboratory, 1977 (La78)

Critical organ and radionuclide	Maximum individual Site boundary (mrem/y)	Maximum individual Nearest resident (mrem/y)	Population[a] (person-rem/y)
Total body			
Tritium	0.42	.09	0.4
Carbon-11, Nitrogen-13 and Oxygen-15	67.0	19.0	7.1
Argon-41	2.1	.9	3.6
Lung			
Plutonium-239	.06	.06	NR
Total	69.5	20	11.1

[a] Total body dose to the population within 80 kilometers.

NR Not reported.

Table 3.4-3. Individual lifetime risks and number of fatal cancers due to radioactive atmospheric emissions from Los Alamos Scientific Laboratory, 1977

Source	Individual lifetime risks			Expected fatal cancers per year of operation (Fatal cancers)[b]
	Site boundary	Nearest resident	Average individual[a]	
LASL	9.7E-4	2.7E-4	1.6E-6	2.2E-3

[a]The region extends to 80 kilometers.
[b]To the population within 80 kilometers.

REFERENCES

DOE77a Department of Energy, 1977, Effluent Information System Reports Nos. 02 and 05, Narrative Summary Data Base Master list and Narrative Summary Data Base Master List Updated Records Report for 1977, (EIS 02 and EIS 05), (Computer Listing).

DOE77b Department of Energy 1977, Effluent Information System Report No. 51, Release Point Analysis Report for Calendar Year 1977 (EIS-51), (Computer Listing).

DOE78 Department of Energy, 1978, Draft Environmental Impact tatement, Los Alamos Scientific Laboratory Site, Los Alamos, New Mexico, DOE/EIS-00/8-D.

LA78 Los Alamos Scientific Laboratory, 1978, Environmental Surveillance at Los Alamos During 1977, UC-41, LA-7263-MS, Los Alamos, N.M.

3.5. Lawrence Livermore Laboratory

3.5.1 General Description

Lawrence Livermore Laboratory (LLL) was established in 1952 for nuclear weapons research and development. In addition to its prime mission, programs are carried out in the areas of magnetic fusion research, non-nuclear energy research, biomedical studies, laser fusion research, and laser isotope separation research.

Lawrence Livermore Laboratory is located about 64 kilometers east of San Francisco, California, on a 254-hectare site in the Livermore Valley of southern Alameda County. The City of Livermore is 5 kilometers to the west of the site.

LLL is situated on an alluvial flood plain bordered by the hills of the Livermore Uplands. The area surrounding the valley is primarily used for pasture lands. The principal agricultural products of the area include grapes and wine, cattle, poultry and eggs. The population within 80 kilometers of LLL is about 4.3 million.

3.5.2 Process Description

There are six principal facilities that release radioactivity into the air at Lawrence Livermore Laboratory.

Light Isotope Handling Facility

Tritium is the principal nuclide released from this facility which is involved with research and development in the area of light isotopes. There is no system employed to reduce tritium from the airborne effluents. The two stacks from this facility are monitored.

Livermore Pool Type Reactor

The Livermore Pool Type Reactor operates at a thermal power of 3 MW to provide neutron irradiations for basic and applied research supporting LLL programs.

Argon-41, produced by air flowing through beam ports and irradiation cells, is essentially the only radioactive stack effluent under normal operating conditions. Argon-41 is released without treatment but the stack is continuously monitored. In the event the reactor building cannot be maintained at negative pressure, air is rerouted through a bank of HEPA filters, activated-charcoal filters, and KOH scrubbers.

Insulated Core Transfer Accelerator (ICT)

The ICT facility houses many medium energy accelerators. However, the ICT accelerator is the only unit having off-site impact.

The ICT accelerator is an air-insulated variable energy machine which accelerates protons and deuterons up to 500 keV. The accelerator uses tritium targets for production of 14 MeV neutrons in support of the Magnetic Fusion Energy Program. Tritium is released from the facility without treatment. The effluent is continuously monitored.

Electron Positron Linear Accelerator (LINAC)

Operation of the 100 MEV LINAC for neutron physics research, produces activation of nitrogen, oxygen, and dust particles in the air of the facility. The activation gases, primarily oxygen-15 and nitrogen-13, are released without treatment. HEPA filters are used to reduce particulate radioactivity in the airborne effluent stream. The effluent stream is continuously monitored before release to the atmosphere from a 30-meter high stack.

Decontamination Facility

HEPA filters are used to reduce particulate radioactivity from exhaust air. The radioactivity in air effluents originate from various decontamination operations. Stack effluents are continuously sampled.

Solid Waste Disposal Facility

Radioactive solid waste packaging, holding, and shipping activities are conducted at this facility. Transfer and compacting operations of dry waste may result in particulate activity being released into the facility ventilation and process air. This air is passed through HEPA filters before release to the atmosphere. During operations the stack effluent is sampled.

3.5.3 Emissions of Radionuclides

Table 3.5-1 identifies radioactive emissions from the facilities at Lawrence Livermore Laboratory in 1977.

Table 3.5-1. Atmospheric emissions of radionuclides, Lawrence Livermore Laboratory, 1977 (DOE77b)

Source	Radionuclide	Emissions (Ci/y)
Light Isotope Handling Facility	Tritium	3.1E+3
Livermore Pool Type Reactor	Argon-41	3.8E+2
ICT Accelerator	Tritium	2.1E+3
Electron Positron Linear Accelerator (LINAC)	Nitrogen-13	5.9E+2
	Oxygen-15	3.9E+2
	Unidentified Beta & Gamma	4.1E-5
Decontamination Facility	Unidentified Beta & Gamma	3.5E-7
Solid Waste Disposal Facility	Unidentified Alpha	2.4E-9
	Unidentified Beta & Gamma	1.1E-8

3.5.4 Health Impact Assessment

Table 3.5-2 summarizes doses from radioactive emissions from Lawrence Livermore Laboratory in 1977. The maximum total body dose to a hypothetical individual at the site boundary of LLL was calculated to be 4.3 mrem from nitrogen-13 and oxygen-15 released from the electron positron linear accelerator. These emissions also resulted in a 1.0 mrem dose to the nearest resident. A population dose of 2.94 person-rem was estimated for the population within 80 kilometers of LLL.

Table 3.5-3 estimates the individual lifetime risks and number of fatal cancers to the population resulting from these doses. The lifetime risk of fatal cancer to the nearest resident is estimated to be 1.4E-5. The number of fatal cancers per year of LLL operation is estimated to be 5.9E-4 to the population within 80 kilometers.

Table 3.5-2. Annual radiation doses from atmospheric emissions of radionuclides from Lawrence Livermore Laboratory, 1977 (Si78)

Source and Radionuclide	Maximum individual[a] Site boundary (mrem/y)	Maximum individual[a] Nearest resident (mrem/y)	Population (Person-rem/y)
Light Isotope Handling Facility			
Tritium	0.3	0.03	0.3
Livermore Pool Type Reactor			
Argon-41	3.2	0.4	2.1
Insulated Core Transfer Accelerator (ICT)			
Tritium	3.0	.04	0.3
Electron Positrom Lincear Accelerator (LINAC)			
Nitrogen-13, Oxygen-15	4.3	1.0	0.2
Total			2.9

[a] Doses are not additive since they occur at different locations.

Table 3.5-3. Individual lifetime risks and number of fatal cancers[a] due to radioactive emissions from Lawrence Livermore Laboratory, 1977

Source	Individual lifetime risks Site boundary	 Nearest resident	 Average individual	Expected fatal cancers per year of operation (Fatal cancers)
LLL	6.0E-5	1.4E-5	9.4E-9	5.9E-4

[a] To the population within 80 kilometers.

REFERENCES

DOE77a Department of Energy, 1977, Effluent Information System Report No. 02, Narrative Summary Data Base Master LIst (EIS 02), (Computer Listing).

DOE77b Department of Energy, 1977, Effluent Information System Report No. 51, Release Point Analysis Report for Calendar Year 1977, (EIS 51), (Computer Listing).

DOE78 Department of Energy, 1978, Draft Environmental Impact Statement, Livermore Site, Livermore, California, DOE/EIS-0028-D.

Si78 Silver W.J., C.L. Lindenken, K.M. Wong, E.H. Willes, J.H. White, 1978, Environmental Monitoring at the Lawrence Livermore Laboratory 1977 Annual Report, Livermore, California.

3.6 Rocky Flats Plant

3.6.1 General Description

The Rocky Flats Plant is part of the national nuclear weapons research, development, and production program. Its primary mission is to produce plutonium components for nuclear weapons.

The Rocky Flats Plant is located in Jefferson County, Colorado, approximately 26 kilometers northwest of Denver. The facilities are located within a 155-hectare security area which is situated on 2650 hectares of Federally-owned land. The site is on the eastern edge of a geological bench, with the foothills of the Rocky Mountains to the west. The area immediately surrounding the plant is primarily agricultural or undeveloped. However, about 1.8 million people reside within 80 kilometers.

3.6.2 Process Description

The Rocky Flats Plant is primarily a radioactive metal fabrication and chemical processing plant. Its mission involves foundry and fabrication of plutonium and uranium components, chemical processing and recovery of plutonium from scrap material, and other transuranic purification operations. These activities are supported by other disciplines such as nuclear safety, engineering, health physics, environmental control, and research and development.

Plutonium at the Rocky Flats Plant is stored in closed containers in a vault with an inert atmosphere. Ingots of plutonium taken from the vault undergo metallurgical processes which include reduction rolling, blanking, forming and heat treating. Smaller pieces of plutonium are drilled or broken to provide samples for the Analytical Laboratory and for casting operations. The formed pieces are then machined into the various components which are then assembled. Assembly operations include cleaning, brazing, marking, welding, weighing, matching, sampling, heating and monitoring. Nuclear weapons are not assembled at this plant.

Solid residue generated during plutonium-related operations are recycled through one of two plutonium recovery processes; the process selected depends on the purity and content of plutonium in the residue. Both processes result in a plutonium nitrate solution from which the metal can be extracted. The recovered plutonium is returned to the storage vault for use in foundry operations. A secondary objective of the process is the recovery of americium-241.

Rocky Flats Plant also conducts operations involving the handling of uranium. Depleted uranium-alloy scrap is consolidated and recycled at one of the foundries. The depleted uranium alloys are ore-melted into ingots for further metallurgical processing. Rocky Flats also has the capabilities to machine and assemble enriched uranium pieces. Enriched uranium components, returned because of age, are disassembled. The enriched uranium is separated and then sent to Oak Ridge, Tennessee, for recycling.

Research and development operations at Rocky Flats Plant are directed towards improving the methods by which plutonium components are produced. Scientific investigations are conducted in all the production oriented fields. These operations are carried out in many areas throughout the plant site.

Because of its toxicity, plutonium is stored and processed under strictly controlled conditions. Much of the plutonium processing equipment is enclosed in glove boxes with an inert, nitrogen atmosphere. The glove boxes are maintained at a slight negative pressure relative to the surrounding area. This allows ventilation air to flow toward areas of greater radioactive contamination instead of away from them.

Many operations, such as handling oxide powder, machining metalic materials, incinerating scrap and waste materials, and chemical recovery processes, have the capability of releasing fine particles of radioactive material into the ventilation air or process gas stream. These effluent streams are passed through HEPA filters to remove the contaminants.

The Rocky Flats Plant uses a minimum of two stages of HEPA filtration for all general building air where plutonium handling is conducted. Plutonium glove box and process air undergoes four stages of HEPA filtration prior to release. Three or four stages of HEPA filters may be expected to provide decontamination factors up to 10^9 to 10^{12}. In buildings where only uranium is handled, at least one stage of HEPA filtration is employed.

3.6.3 Emissions of Radionuclides

Since the upgrading of ventilation and HEPA filtration systems at the plutonium processing facilities was completed in 1970, plutonium releases to the atmosphere have been held below 100 microcuries per year from normal operations. Based on current performance, the expected yearly atmospheric emissions of plutonium from Rocky Flats would be about 10 microcuries per year. Table 3.6.1 shows that the 1977 releases were below that level.

Following improvements made to the filtration systems in the uranium handling buildings in 1970, yearly releases of depleted and enriched uranium have been less than 50 microcuries per year each.

The tritium released in 1977 from Rocky Flats was due to residuals from a release in 1973 when tritium contaminated material was inadvertently processed.

Table 3.6-1. Atmospheric emissions of radioactive materials from the Rocky Flats Plant, 1977 (DOE77b)

Radionuclide	Emissions (Ci/y)
Tritium	5.3E-1
Plutonium-239 and Plutonium-240	4.2E-6
Uranium 235	2.1E-5
Uranium-238	1.9E-5

3.6.4 Health Impact Assessment of Rocky Flats Plant

The estimated doses resulting from radioactive emissions from the Rocky Flats Plant in 1977 are listed in table 3.6-2. The maximum potential dose at the site boundary, based on continuous exposure to measured concentrations of plutonium in the air, was to the east of the plant. The dose of about 0.2 millirem to a hypothetical resident of Denver in 1977 was based on exposure to concentrations of plutonium in air in the Denver area in excess of background concentrations. The Rocky Flats population dose was based on atmospheric concentrations measured in surrounding communities. The total dose to the population within 80 kilometers was about 200 person-rems.

Table 3.6-3 estimates the individual lifetime risks and fatal cancers in the population resulting from these doses. The number of fatal cancers for plant operation in 1977 is estimated to be about 7.2E-3 to the population within 80 kilometers.

Table 3.6-2. Annual radiation doses[a] from radionuclides from Rocky Flats Plant, 1977

Organ	Maximum individual Site boundary (mrem/y)	 Denver resident (mrem/y)	Population[b] (Person-rem/y)
Lung	8.0E-2	1.3E-1	1.3E+2
Bone	3.5E-2	5.5E-2	5.8E+1
Kidney	4.0E-3	6.3E-3	6.6
Liver	5.4E-3	8.5E-3	8.9
Total body	8.6E-4	1.4E-3	1.4

[a] Doses are based on measured concentrations in the air in excess of background concentrations. Denver concentrations are higher than site boundary concentrations.

[b] The population is residing within 80 kilometer of the plant.

Table 3.6-3. Individual lifetime risks and number of fatal cancers[a] due to radioactive emissions from Rocky Flats Plant, 1977

Organ	Individual lifetime risks Maximum individual at site boundary	 Denver resident	Expected fatal cancers per year of operation (Fatal cancers)
Lung	2.2E-7	3.5E-7	5.3E-3
Bone	7.4E-8	1.2E-7	1.7E-3
Kidney	2.8E-9	4.4E-9	6.6E-5
Liver	3.8E-9	5.9E-9	8.9E-5
Other soft tissue	3.0E-9	4.7E-9	7.1E-5
Total	3.0E-7	4.9E-7	7.2E-3

[a] To the population within 80 kilometers.

REFERENCES

DOE77a Department of Energy, 1977, Effluent Information System Report Nos. 02 and 05, Narrative Summary Data Base Master list and Narrative Summary Data Base Master List Updated Report for 1977, (EIS 02 and EIS 05), (Computer Listing).

DOE77b Department of Energy, 1977, Effluent Information System Report No. 50, Release Point Analysis Report (EIS 51), (Computer Listing).

Ro78 Rockwell International, 1978, Annual Environmental Monitoring Report, January-December 1977, RFP-ENV-77, Rocky Flats, Colorado.

3.7 Mound Facility

3.7.1 General Description

The Mound Facility has been in operation since 1949. Its primary functions include research, development, engineering, production and surveillance of components for the DOE weapons program. Other operations involve the separation, purification, and sale of stable noble gas isotopes and the fabrication of radioisotopic heat sources for thermoelectric generators.

The Mound facility, located in Miamisburg, Ohio, about 16 kilometers southwest of Dayton, occupies a 73-hectare site in the Great Miami River Valley. This area is highly industrialized. The surrounding region is mostly agricultural with some light industry and scattered residential communities. About 2.8 million people live within 80 kilometers of the Mound Facility.

3.7.2 Process Description

Nine buildings at the Mound Facility released radioactivity into the atmosphere in 1977. Operations at these facilities resulted in the release of tritium and plutonium-238.

Tritium was released in atmospheric effluents from the HH and SW buildings. Operations at the HH building involve the recovery of helium-3 which is contaminated with tritium. Gaseous wastes generated here are stored and transfered to the SW building. At the SW building operations involve disassembly, analysis and development of nuclear components containing tritium, and the recovery of tritium wastes. Tritium in gaseous effluents streams of the SW building are treated before release by the effluent removal system, which oxidizes elemental tritium and then removes the resulting tritiated water by molecular sieve drying beds.

Plutonium-238 was released in airborne effluents from PP, R, WD, WDA, 41, H and SM buildings. Plutonium processing and other related activities are conducted at the PP building. At the R building plutonium heat source production is the principal activity. Operations at the WD, WDA and 41 buildings involve radioactive waste disposal processes. Contaminated clothing is laundered at the H building, and the SM building is an idle contaminated facility. At all these facilities, particulate radioactivity is removed from process air streams by HEPA filters.

The airborne effluents undergo filtration at their point of generation and again at the stack just prior to release.

3.7.3 Emissions of Radionuclides

In 1977 a total of 4.9E+3 curies of tritium and 0.1 microcuries of plutonium-238 were released from the Mound Facility. Table 3.7.1 shows the atmospheric emissions from each building.

Table 3.7-1. Emissions of radionuclides from the Mound Facility, 1977 (DOE77b)

Facility	Radionuclide	Emissions (Ci/y)
H Building	Plutonium-238	4.9E-13
HH Building	Tritium	9.5E+01
PP Building	Plutonium-238	4.1E-06
R Building	Plutonium-238	3.7E-07
SM Building	Plutonium-238	4.6E-06
SW Building	Tritium	4.8E+03
WD Building	Plutonium-238	3.7E-08
WDA Building	Plutonium-238	1.7E-06
41 Building	Plutonium-238	1.2E-07

3.7.4 Health Impact Assessment of the Mound Facility

The maximum individual annual dose to a person at the site boundary of the Mound Facility due to 1977 radioactive atmospheric emissions was 0.11 rem/y to the lung from inhalation of plutonium-238. These same releases resulted in a maximum dose to the lung of .04 mrem/y to the nearest resident. Dose estimates to the bone and whole body from exposure to atmospheric releases of plutonium and tritium from the Mound Facility are shown in Table 3.7-2. The maximum lifetime risk of fatal cancer to the nearest person is 2.8E-7.

The dose to the population within 80 kilometers from airborne tritium was calculated to be 3.7 person-rem for 1977. This would correspond to 7.2E-4 fatal cancers per year of operation to the population surrounding the Mound Facility.

Table 3.7-2. Annual radiation doses from atmospheric emissions, Mound Facility, 1977 (Fa77)

Organ	Maximum individual[a] Site boundary (mrem/y)	Maximum individual[a] Nearest resident (mrem/y)	Population (Person-rem/y)
Bone	4.5E-2	1.5E-2	NR
Lung	1.1E-1	4.0E-2	NR
Total body	4.0E-2	2.0E-2	3.6

[a] Doses are not additive since they occur at different locations.
NR Not reported.

Table 3.7-3. Individual lifetime risks and number of fatal cancers due to radioactive emissions from Mound Facility, 1977

Organ	Individual lifetime risks Nearest resident	Individual lifetime risks Average individual Region	Expected fatal cancers per year of operation (Fatal cancers)
Bone	3.2E-8	NR	NR
Lung	1.0E-7	NR	NR
Total body	2.8E-7	1.8E-8	7.2E-4

NR Not reported.

REFERENCES

DOE77a Department of Energy, 1977, Effluent Information System Report Nos. 02 and 05, Narrative Summary Data Base Master List and Narrative Summary Data Base Master List Updated Report for 1977, (EIS 02 and EIS 05).

DOE77b Department of Energy, 1977, Effluent Information System Report No. 51, Release Point Analysis Report for Calendar Year 1977 (EIS 51).

Fa78 Farmer B., Robinson B. M., Carfagno D.G., 1978, Annual Environmental Monitoring Report: Calendar Year 1977, MLM-2515.

3.8 Pantex Plant

3.8.1 General Description

The Pantex Plant's mission includes atomic weapons assembly, retirement, and stockpile surveillance. The plant also fabricates and tests chemical explosives.

The Pantex Plant, located on 3683 hectares of land in Carson County, Texas, is in the Panhandle Plains area about 27 kilometers northeast of Amarillo. The Panhandle is predominately an agricultural area; the population is about 236,000 persons within 80 kilometers.

3.8.2 Process Description

High explosive test firings, which are conducted intermittently, release small quantities of depleted uranium. Disassembly, shipping, and receiving operations within the plant release small amounts of tritium. There are no sources of continuous releases of radioactive material to the atmosphere. With few exceptions, radioactive material is handled in sealed containers. These containers are not opened, thus avoiding the possiblity of release during normal operations. There are no systems employed to treat the radioactive emissions to the atmosphere.

3.8.3 Emissions of Radionuclides

The estimated release of radioactive materials for 1977 from Pantex Plant are shown in Table 3.8-1.

Table 3.8-1. Atmospheric emissions of radionuclides from the Pantex Plant, 1977 (A178)

Facility	Radionuclide	Emissions (Ci/y)
Firing site	Uranium-238	1.0E-3
Assembly area	Tritium	1.0E-2

3.8.4 Health Impact Assessment of the Pantex Plant

Table 3.8-2 summarizes the estimated radiation doses due to radioactive emissions from the Pantex Plant. The principal radionuclide contibuting to these doses is uranium-238.

The total body dose to the the population within 80 kilometers of the Pantex Plant in 1977 was 4.0E-4 person-rem from uranium-238 and 2.0E-7 person-rem from tritium. These doses would result in an estimated 8.0E-8 fatal cancers per year of plant operations to the population within 80 kilometers. Table 3.8-3 summarizes the individual lifetime risks and health effects associated with Pantex Plant operations.

Table 3.8-2. Annual radiation doses due to radioactive emissions from Pantex Plant, 1977 (A178)

Organ	Maximum individual Site boundary (mrem/y)	Nearest resident (mrem/y)	Population (person-rem/y)
Lung	9.0E-5	6.0E-5	NR
Kidney	2.0E-4	1.0E-4	NR
Total body	3.0E-6	2.0E-6	4.0E-4

NR Not reported.

Table 3.8-3. Individual lifetime risks and number of fatal cancers[a] from Pantex Plant, 1977

Organ	Individual lifetime risks Nearest resident	Average individual Region	Expected fatal cancers per year of operation (Fatal cancers)
Lung	1.7E-10	NR	NR
Kidney	7.0E-11	NR	NR
Other soft tissue	7.0E-12	NR	NR
Total	2.5E-10	2.4E-11	8.0E-8

[a] To the population within 80 kilometers.
NR Not reported.

REFERENCES

A178 Alexander Ronald E., and C. Newlyn Horton, 1978, Environmental Monitoring Report for Pantex Plant Covering 1977, MHSMP-78-7.

DOE77a Department of Energy, 1977, Effluent Information System Reports Nos. 02 and 05. Narrative Summary Data Base Master List and Narrative Summary Data Base Master List Updated Report for 1977, (EIS 02 and EIS 05), (Computer Printout).

DOE77b Department of Energy, 1977, Effluent Information System Report No. 51, Release Point Analysis Report For Calendar Year 1977, (EIS 51), (Computer Printout).

3.9 Pinellas Plant

3.9.1 General Description

Pinellas Plant is operated by the Neutron Devices Department of the General Electric Company. Operations involve the design, development, and manufacture of special electronic and mechanical components for nuclear weapons.

Pinellas Plant, located on a 39.2 hectare-tract in Pinellas County, Florida, is on Florida's West Coast, north of St. Petersburg. The county has had a rapid population growth and is the most densely populated county in Florida with about 1.7 million people residing within 80 kilometers of the plant.

3.9.2 Process Description

The principal operations causing atmospheric releases of radioactive materials are not described in the literature. However, they involve neutron generator development and production, testing, and laboratory operations. Areas utilizing radioactive materials are connected to a special exhaust system which is designed to trap tritium and reduce the amount released to the atmosphere. In this system tritium gas is converted to the oxide form by passage through heated copper oxide beds. Then the tritiated water vapor is absorbed by silical gel.

Small sealed plutonium capsules are used as heat sources in the manufacture of radioisotopic thermoelectric generators at Pinellas Plant. These sources are triply encapsulated so as to prevent release of plutonium to the atmosphere.

3.9.3 Emissions of Radionuclides

Small amounts of tritium gas, tritium oxide, and krypton-85 were released from one of two 30-meter stacks at Building 100. The releases for 1977 are summarized in Table 3.9-1.

3.9.4 Health Impact Assessment of Pinellas Plant

Table 3.9-2 summarizes doses from radioactive atmospheric emissions from the Pinellas Plant in 1977. The maximum site boundary dose occurred along the western perimeter while the nearest residential area is located approximately 1.6 kilometers to the south-southeast of the plant. A population dose of 0.4 person-rem from atmospheric emissions of tritium was estimated for the nearly 1.7 million people living within 80 kilometers of the Pinellas Plant.

Estimates of the individual lifetime risks and the number of fatal cancers resulting from these doses are given in table 3.9-3.

Table 3.9-1. Atmospheric emissions of radionuclides from Pinellas Plant, 1977 (DOE77b)

Source	Radionuclide	Emissions (Ci/y)
Building 100-Main stack	Tritium Krypton-85	1.6E+2 2.8E+1
Building 100-Laboratory stack	Tritium	1.3E+2

Table 3.9-2. Annual radiation doses from atmospheric emissions of radioactive materials from Pinellas Plant, 1977 (GE78)

Radionuclide	Maximum individual Site boundary (mrem/y)	Maximum individual Nearest resident (mrem/y)	Population[a] (person-rem/y)
Krypton-85	1.3E-3	NR	NR
Tritium	1.0E-2	3.8E-3	NR
Total	1.1E-2	3.8E-3	0.4

[a] The population is within 80 kilometers.
NR Not reported.

Table 3.9-3. Individual lifetime risks and number of fatal cancers resulting from atmospheric emissions of radioactive materials from Pinellas Plant, 1977

Radio-nuclide	Individual lifetime risks			Expected fatal cancers per year of operation (Fatal cancers)
	Maximum Individual	Nearest Resident	Average individual	
Krypton-85	1.8E-8	NR	NR	NR
Tritium	1.4E-7	5.3E-8	3.3E-9	8.0E-5

[a] To the population within 80 kilometers.

NR Not reported.

REFERENCES

DOE77a Department of Energy, 1977, Effluent Information System Reports Nos. 02 and 05, Narrative Summary Data Base Master List and Narrative Summary Data Base Master List Updated Report for 1977, (EIS 02 and EIS 05), (Computer Printout).

DOE77b Department of Energy, 1977, Effluent Information System Report No. 51, Release Point Analysis Report for Calendar Year 1977, (EIS 51), (Computer Printout).

GE78 General Electric Company, 1978, Pinellas Plant Environmental Monitoring Report, St. Petersburg, Florida.

Rh73 Rhinehammer T.B. and P.H. Lamberger, 1973, Tritium Control Technology, WASH-1269, U.S. Atomic Energy Commission, Miamisburg, Ohio.

3.10 Sandia Laboratories

3.10.1 General Description

Sandia Laboratories is a nuclear ordnance laboratory which combines nuclear weapons developed by Los Angeles Scientific Laboratory and Lawrence Livermore Laboratory with delivery systems needed by the military services. This responsibility includes performing weapons testing, quality control and assurance, arming and fusing, safety, delivery system modification, and safeguards. Components are tested for proper operation under a variety of environmental conditions involving parameters such as shock, vibration, temperature, moisture and radiation.

The Sandia Laboratories, are located in Albuquerque, New Mexico, and Livermore, California. The Livermore site is adjacent to Lawrence Livermore Laboratory. In Albuquerque, Technical Area V is where much of the radiation testing is performed. This area is located approximately 10 kilometers south of the city in a sparsely populated region--about 380,000 people live within 80 kilometers, mostly inside a 20-kilometer radius.

3.10.2 Process Description

Albuquerque Site

As part of the environmental testing capability, Sandia Laboratories operates two research reactors which are located in Technical Area V. The Sandia Pulsed Reactor is an unreflected, cylindrical, enriched-uranium assembly. The Annular Core Pulse Reactor is a modified TRIGA-type reactor. Reactor operations release small amounts of fission and activation product gases, primarily argon-41. Both reactor building exhaust effluents undergo HEPA filtration prior to monitoring and release.

Also located within Technical Area V is the Relativistic Electron Beam Accelerator Facility which is used for electron beam fusion research. Small amounts of tritium gas are emitted from the vacuum pump exhaust. Tritium gas in the effluent stream is catalyzed to the oxide from which it is then removed by molecular sieves.

Technical Area I has research, design, administrative and support facilities. The principal operation emitting radioactivity to the atmosphere in this area is from neutron activation experiments. Small quantities of tritium are released

Table 3.10-1. Atmospheric emissions of radionuclides from Sandia Laboratories, 1977 (DOE77b)

Source	Radionuclide	Emissions (Ci/y)
Albuquerque Site		
Annular Core Pulsed Reactor	Argon-41	3.9
Sandia Pulsed Reactor	Argon-41	9.2E-01
Relativistic Electron Beam Accelerator	Tritium	4.0E-07
Neutron Generator	Tritium	1.0E-02
Livermore Site		
Building 913	Unidentified Alpha	1.8E-10
	Unidentified Beta & Gamma	1.8E-09
	Tritium	8.2E-01
Building 968	Tritium	1.0E-02

from a neutron generator housed in building 805. The building ventilation system is not equipped with any radioactive waste treatment system.

Livermore Site

Laboratories at the Livermore Site include those for electronics, telemetry, nucleonics, optical electronics, powder metallurgy, hydrogen effects, and microelectronics.

Two buildings at this site release radioactive airborne effluents, building 913 and building 968. In building 913 assembly and dissasembly of radioactive materials are conducted along with cutting and polishing operations. The process air from these operations is passed through HEPA filters before being released and the stacks are sampled for alpha and beta radiation.

Building 968, the Tritium Research Laboratory, is designed for experiments involving kilocurie quantities of tritium. The facility uses containment and cleanup capabilities. Tritium from glovebox operations and vacuum pump exhaust is removed by a gas purification system. This is accomplished by catalytic or chemical oxidation of the tritium gas to the water form, then collecting it on a molecular sieve bed.

3.10.3 Emissions of Radionuclides

Radioactive releases to the atmosphere from Sandia include argon-41 from Area V reactors and tritium from research activities in both Areas I and V. Table 3.10-1 summarizes the releases for 1977.

3.10.4 Health Impact Assessment of Sandia Laboratories

The maximum dose to a hypothetical individual at the Albuquerque site boundary from atmospheric releases of radioactivity in 1977 was 2.0E-3 millirem from tritium releases from Technical Area I, and 1.0E-3 millirem from argon-41 released from Technical Area V. These doses correspond to an estimated lifetime risk of fatal cancer of 2.8E-8 and 1.4E-8, respectively.

The dose to the population within 80 kilometers from the 1977 airborne effluents from the Albuquerque site was 5.5E-2 person-rem, primarily from argon-41. An estimated 1.1E-5 fatal cancers would result from each year of operations at the site.

The radioactive atmospheric emissions for the Sandia Laboratories, Livermore Site, are small when compared to the emissions for the Lawrence Livermore Laboratory. They would not contribute to any significant increase to that dose resulting from Lawrence Livermore Laboratory operations which is reported in Section 3.5 of this report.

Table 3.10-2. Annual radiation doses from radioactive emissions at Sandia Laboratories, 1977

Source	Maximum individual Site boundary (mrem/y)	Population[a] (Person-rem/y)
Albuquerque Site:		
Technical Area I		
Tritium	2.0E-3	2.2E-5
Technical Area V		
Argon-41	1.0E-3	5.5E-2

[a] The population is within 80 kilometers.

Table 3.10-3. Individual lifetime risks and number of fatal cancers from radioactive emissions, Sandia Laboratories, 1977

Source	Individual lifetime risks[a]		Expected fatal cancers per year of operation[b] (Fatal cancers)
	Maximum individual	Average individual in the region	
Tritium	2.8E-8	1.2E-14	4.4E-9
Argon-41	1.4E-8	3.5E-11	1.1E-5

[a] The maximum individual is at the site boundary.
[b] The population is within 80 kilometers.

REFERENCES

DOE77a Department of Energy, 1977, Effluent Information System Report Nos. 02 and 05, Narrative Summary Data Base Master List and Narrative Summary Data Base Master List Updated Reports for 1977. (EIS 02 and EIS 05), (Computer Printout).

DOE78 Department of Energy, 1978, Draft Environmental Impact Statement, Livermore Site, Livermore, California, DOE/EIS-0028-D.

DOE77b Department of Energy, 1977, Effluent Information System Report No. 51, Release Point Analysis Report (EIS 51), (Computer Printout).

Si78 Simmons Theodore N., 1978, Environmental Monitoring Report Sandia Laboratories, 1977, SAND78-0620.

3.11 Nevada Test Site

3.11.1 General Description

The Nevada Test Site (NTS) is a part of the National weapons research and development program. Nuclear weapons testing and experiments on the site are performed in conjunction with weapons systems developed at Lawrence Livermore Laboratory, the Los Alamos Scientific Laboratory, and the Sandia Laboratories in response to the Department of Defense requirements.

The Nevada Test Site is located in Nye County, Nevada, about 100 kilometers northwest of Las Vegas. It occupies about 349,000 hectares of Federally-owned land. The Nellis Air Force Base and Tonopah Test Range border the Nevada Test Site on three sides, from the northwest to the east, providing an additional one million hectares of Federally-owned land as a buffer. With the exception of Las Vegas, the region surrounding the NTS is rural and sparsely populated--only 4,500 people live within 80 kilometers of NTS. Beatty, with a population of 500, is the largest town nearby.

3.11.2 Process Description

All nuclear weapons detonations at the NTS, have been conducted underground since the Limited Test Ban Treaty in 1963. As of January 1, 1977, there had been 289 announced underground nuclear tests. Underground detonations contain the large amounts of radioactive material created in the cavity formed by the explosion. Since 1971, there have been no prompt ventings or inadvertent releases of gaseous radioactivity from test explosions. After the test detonations, one or more re-entry holes are drilled back into the radioactive debris for samples to determine the performance of the device.

During sample recovery operations, small quantities of volatile radionuclides, primarily xenon-133, are brought to the surface and released to the atmosphere. In order to minimize releases, a system is used to either force the effluent gases back into the drill hole or to pass the effluent through prefilters, charcoal filters, and HEPA filters prior to release to the atmosphere.

3.11.3 Emissions of Radionuclides

During 1977, re-entry drilling operations resulted in occasional low-level releases of airborne radioactivity, primarily radioxenon. There was also some small leakage of tritium to the atmosphere from the waste tritium storage area. Table 3.11-1 details the quantities of radionuclides released to the atmosphere in 1977.

Continuous low-level releases of tritium and krypton-85 occur at NTS. These radionuclides may seep to the surface from the sites of underground testing. Tritium in drainage ponds is released by evaporation; the amounts from seepage and evaporation are not quantified, but are detected at on-site sampling stations and sometimes at locations outside of the NTS.

Table 3.11-1. Atmospheric emissions of radionuclides, Nevada Test Site, 1977 (DOE77b)

Radionuclide	Emission (Ci/y)
Tritium	4.1
Iodine-131	2.6E-6
Xenon-133	4.6E+1
Xenon-133m	6.2E-1
Xenon-135	8.5E-1

3.11.4 Health Impact Assessment of Nevada Test Site

The only radionuclide attributed to NTS operations was xenon-133 which was detected at Beatty, Diablo, Hiko, Las Vegas and Tonopah, Nevada. The highest levels detected resulted in a maximum annual whole-body dose of 2.5 microrem at Beatty (table 3.11-2).

The maximum individual lifetime risk of fatal cancer is estimated to be about 3.5E-8. There would be about 7.2E-5 fatal cancers per year of operation to the populations listed in Table 3.11-2.

Table 3.11-2. Estimated annual radiation doses from xenon-133 concentrations, 1977 (EPA78)

Location	Population	Dose equivalent (microrem/y)	Population dose commitment (person-rem/y)	Population dose commitment[a] (person-rem/y)
Beatty, NV	500	2.5	1.3E-3	1.3E-3
Diablo, NV	6	1.2	7.2E-6	0.0
Hiko, NV	60	1.1	6.6E-5	0.0
Las Vegas, NV	[b]370,500	1.0	3.6E-1	0.0
Tonopah, NV	2,000	1.4	2.8E-3	0.0
Total			3.6E-1	1.3E-3

[a] To the population within 80 kilometers of NTS.
[b] Las Vegas and nearby communities within Clark County.

Table 3.11-3. Individual lifetime risks and number of fatal cancers from Nevada Test Site operations, 1977

Radio- nuclide	Individual lifetime risks Nearby resident in Beatty, NV	Individual lifetime risks Average individual[a]	Expected fatal cancers per year of operation[a] (Fatal cancers)
Xenon-133	3.5E-8	3.6E-8	7.2E-5

[a] To the population at the locations listed in table 3.11-2.

REFERENCES

DOE77a Department of Energy, 1977, Effluent Information System Report Nos. 02 and 05, Narrative Summary Data Base Master List and Narrative Summary Data Base Master List update Records Report 1977, (EIS 02 and EIS 05).

DOE77b Department of Energy, 1977, Effluent Information System Report No. 51, Release Point Analysis Report for Calendar Year 1977, (EIS 51).

EPA78 Environmental Protection Agency, 1978, Off-Site Environmental Monitoring Report for The Nevada Test Site and Other Test Areas Used for Underground Nuclear Detonations January through December 1977.

ERDA77 Environmental Research and Development Administration, 1977, Final Environmental Impact Statement, Nevada Test Site, Nye County, Nevada, ERDA-1551.

3.12 Argonne National Laboratory

3.12.1 General Description

Argonne National Laboratory (ANL) is a multidisciplinary research and development laboratory conducting a broad program of research in the physical, biomedical and environmental sciences. It is also an important center for nuclear and non-nuclear energy research and development.

Argonne National Laboratory is located in Dupage County, Illinois, about 43 kilometers southwest of downtown Chicago. The laboratory facilities are located on the central portion of a 1513-hectare site. Approximately 8.1 million people live within 80 kilometers of ANL.

3.12.2 Process Description

The principal sources of atmospheric emissions of radioactive materials from ANL are the CP-5 and Janus research reactors. Other nuclear facilities of ANL include a critical assembly or zero power reactor, the Argonne Thermal Source Reactor, a 12.5 GeV proton accelerator, 60-inch cyclotron, several other particle accelerators, chemical and metallurgical plutonium laboratories, and several hot cells and laboratories designed for work with irradiated fuel elements.

The CP-5 reactor is a 5 MW fully-enriched heavy water, general purpose research reactor. It uses a helium cover gas system to provide an inert atmosphere for the reactor and to maintain isotopic purity of the D_2O in the primary system. Neutron activation of room cooling air and tritium leaks are the main sources of effluents from the CP-5 reactor. These effluents are released without treatment.

The Janus reactor is a 200 KW light water biological research reactor which is fueled with fully enriched uranium. Neutron activation of the atmosphere in the high dose room is the source of radioactivity released in effluents from this facility. No waste treatment system is in use for radioactive airborne effluents at this facility.

The hot cells, where nuclear reactor material examinations are conducted, are equipped with two stages of HEPA filters. These HEPA filters are not effective in reducing radioactive gases that may be in the effluent from these cells.

At one building, where uranium grinding activities are conducted, the airborne effluent stream is treated with a wet scrubber and a electrostatic precipitator.

3.12.3 Emissions of Radionuclides

Argon-41 from reactor operations was the principal radionuclide contributing to radioactive emissions to the atmosphere from Argonne National Laboratory in 1977. Table 3.12-1 shows the quantities released in 1977 from ANL.

Table 3.12-1. Atmospheric emissions of radionuclides from Argonne National Laboratory, 1977 (DOE77b)

Source	Radionuclide	Emissions (Ci/y)
CP-5 Reactor	Argon-41	3.0E+4
	Tritium	8.5E+2
Janus Reactor	Argon-41	3.4
Hot Cells	Krypton-85	1.4E+1
	Plutonium-239	1.3E-8
	Antimony-125	4.9E-5
Various laboratoriess	Tritium	6.5E-1
	Krypton-85	2.0E-1
	Uranium-238	1.9E-7

3.12.5 Health Impact Assessment of Argonne National Laboratory

The maximum doses from airborne emissions in 1977 occurred to the northeast of Argonne National Laboratory. The maximum site boundary doses from argon-41 and tritium were 5.6 millirem and 0.01 millirem; the dose to the nearest fulltime resident was 3.0 millirem. The effluents from the CP-5 Reactor resulted in a population dose for 1977 of 178 person-rem.

Table 3.12-2. Annual radiation doses from radioactive emissions from the CP-5 reactor at ANL (Go78)

Radionuclide	Site boundary dose[a] (mrem/y)	Nearest resident[a] (mrem/y)	Population dose[b] (person-rem/y)
Argon-41	5.6	3.0	177
Tritium	1.0E-2	7.0E-3	1.1
Total			178

[a] These doses are not additive since they occur at different locations.

[b] Within 80 kilometers.

Table 3.12-3 estimates the individual lifetime risks and the number of fatal cancers resulting from these doses. The lifetime cancer risks to the maximum individual at the site boundary and to the nearest fulltime resident are estimated to be 7.8E-5 and 4.2E-5, respectively. The estimated number of fatal cancers to the population within 80 kilometers per year of ANL operation is estimated to be about 3.6E-2.

Table 3.12-3. Individual lifetime risks and number of fatal cancers[a] from radioactive emissions from the CP-5 reactor at ANL, 1977

Source	Individual lifetime risks			Expected fatal cancers per year of operation (Fatal cancers)
	Site boundary	Nearest fulltime resident	Average individual	
CP-5 reactor	7.8E-5	4.2E-5	3.1E-7	3.6E-2

[a] To the population within kilometers.

REFERENCES

DOE77a U.S. Department of Energy, 1977, Effluent Information System Report Nos. 02 and 05, Narrative Summary Data Base Master List and Narrative Summary Update Records Report for Calendar Year 1977, (EIS 02 and EIS 05).

DOE77b U.S. Department of Energy, 1977, Effluent Information System Report No. 51, Release Point Analysis Report for Calendar Year 1977, (EIS 51).

Go78 Golchert, N.W., T.L. Duffy, and J. Sedlet, 1978, Environmental Monitoring At Argonne National Laboratory, Annual Report for 1977, ANL-78-26.

3.13 Brookhaven National Laboratory

3.13.1 General Description

Brookhaven National Laboratory (BNL) is a multidisciplinary scientific research center which supports a wide variety of scientific, research, and development programs.

The laboratory facilities are located at Upton, in Suffolk County, New York, about 113 kilometers east of New York City. The BNL facilities occupy the central 405 hectares of a wooded 2130-hectare site. The principal population centers, located along the shoreline of Long Island, are within 16 kilometers. Approximately 5.1 million people live within 80 kilometers of Brookhaven.

3.13.2 Process Description

Among the major facilities at Brookhaven National Laboratory are the High Flux Beam Reactor (HFBR), the Tandem Van de Graaff Accelerator, the Brookhaven Medical Research Reactor (BMRR), the 200 MeV Proton Linear Accelerator (Linac), which is operated in conjunction with the Alternating Gradient Synchrotron (AGS) and the Brookhaven Linac Isotope Producer Facility (BLIP).

The High Flux Beam Reactor (HFBR) is a 40 MW heavy water fully enriched, uranium research reactor. It is used to provide intense beams of neutrons for research. The HFBR has an inert cover gas system to maintain purity of the heavy water coolant. At the end of each operating cycle (10 times per year), the purging of the cover gas releases tritium-bearing effluents from the 98-meter HFBR stack. Tritium is also released from valve and pump seals from evaporation during refueling. Airborne radioactivity produced during routine operations is vented through HEPA and activated charcoal filters before being discharged from the stack.

The hot area of the Hot Laboratory consists of five semihot cells, three chemical processing hot cells and three high-level hot cells for handling multicurie amounts of radioactive materials. Each cell is equipped with its own exhaust air filter as well as a backup HEPA filter in the exhaust line leading to the stack. The process cells have a separate exhaust air system which use a NaOH scrubber and charcoal filter to remove radioiodines. The hot cells exhaust the airborne effluents to the HFBR stack.

The Tandem Van de Graaff accelerator started operations in 1970. Each Van de Graaff electrostatic accelerator can accelerate atomic particles up to 10 MeV. The maximum combined energy that can be achieved is 30 MeV. Accelerator produced radiation in the ventilation air includes trace quantities of the short-lived gases carbon-11, nitrogen-13 and oxygen-15. The Tandem Van de Graaff accelerator building air is vented to the HFBR stack. Concentrations of radioactive gases from the accelerator are not detectable by the stack gas monitor.

The Medical Research Reactor is a 5 MW tank-type reactor which uses 12 weight percent fully-enriched uranium fuel. The water-cooled reactor core is within an aluminum vessel surrounded by an air-cooled graphite reflector and biological shield. The upper surface of the primary cooling water in the reactor vessel is in contact with the air cooling the reflector. The cooling air picks up gaseous contaminants from the water. These gases along with the neutron activated gas, argon-41, are exhausted through HEPA and charcoal filters and released from a 46-meter stack.

The Alternating Gradient Synchrotron (AGS) accelerates protons up to 33 GeV. This accelerator is used for ultra-high energy particle physics research. Protons for the AGS system originate from a Cockcroft-Walton generator which gives the protons an initial energy of 750 KeV. These protons are then injected into a linear accelerator (Linac) which accelerates them up to 200 MeV. The proton beam is then injected into the 0.8 kilometer circular path of the AGS vacuum chamber. The resultant beam can then be bent to strike a target or deflected out of the ring into experimental areas. Because the proton beam is highly focused and in a vacuum, there is minimal activation of air in the surrounding tube. Carbon-11, nitrogen-13 and oxygen-15 are the predominant nuclides produced. Release of these short-lived activation gases is minimal since most of the tunnel air is recirculated.

The 200 MeV Linac is also used in conjunction with the Brookhaven Linac Isotope Producer Facility (BLIP) and the infrequently used Chemistry Linac Irradiation Facility (CLIF). Targets for radionuclide production at the BLIP facility are irradiated at the bottom of a 10-meter, 2.4-meter diameter, water-filled tank. The targets are sealed to prevent escape of radioactivity during normal operations. Several radioactive gases are induced by the incident protons in the target cooling water. These include tritium, nitrogen-13, oxygen-14, oxygen-15 and nitrogen-16. The radioactive oxygens have larger release rates in relation to production rates than do the other radioactive gases because they are swept out with the absorbed oxygen in the cooling water by the radiolytic formation of stable oxygen. Several radiocarbons are formed as a result of proton interaction with the water. However, release of these radiocarbons as gaseous carbon dioxide is minimized by keeping the cooling water alkaline with NaOH. The airborne effluents from the BLIP Facility undergo HEPA filtration prior to monitoring and release from an 18-meter stack.

A number of lower energy accelerators are used for medium energy physics experiments at BNL. These accelerators include a 3 MeV Dynamitron (electron accelerator), a 60-inch cyclotron (30 MeV), the Febetron (1.9-MeV pulsed accelerator) and three Van de Graaff accelerators which range in energy from 2 to 6 MeV. The principal atmospheric emissions come from the physics research Van de Graaff which is used to accelerate tritium to 3.5 MeV. Effluent from the vacuum system is passed through the scrubber where tritium gas is converted to the oxide which is then trapped in a desiccant. When the system is first pumped down, the flow exceeds the capacity of the recombiner and is routed to a bypass. Fifty percent of the tritium used is trapped by the desiccant while the remainder is released from an 18-meter stack. The other accelerators are operated only intermittently and produce short-lived gases which are released with room ventilation air at roof level.

In addition to the major facilities already mentioned, there are some 432 laboratory hoods located within various buildings at Brookhaven. These hoods discharge air through pipes to vents on the roof. Those hoods, where toxic agents and millicurie amounts of radionuclides are handled, are equipped with HEPA filters or scrubbers to control emissions.

3.13.3 Emissions of Radionuclides

The principal releases from Brookhaven National Laboratory are tritium and argon-41 from the reactors along with tritium and short-lived activation gases from accelerator operations. Tritium is the only airborne effluent detectable off-site. Table 3.13-1 shows the radioactive atmospheric releases from BNL in 1977.

Table 3.13-1. Atmospheric emissions from radionuclides, Brookhaven National Laboratory, 1977 (DOE77b)

Source	Radionuclide	Emissions (Ci/y)
High Flux Beam Reactor	Tritium	1.2E+2
	Unidentified Beta & Gamma	1.1E-5
	Xenon-127	1.1
Medical Research Center		
Building 490	Tritium	1.4
Medical Reactor	Argon-41	3.6E+2
Research Van de Graaff Accelerator	Tritium	1.0E+3
Linac Isotope Producer Facility (BLIP)	Tritium	8.4E-2
	Oxygen-15	6.7E+4
Chemistry Building-555	Tritium	7.1E+1

3.13.4 Health Impact Assessment of Brookhaven National Laboratory

Oxygen-15 and argon-41 are released in significant quantities from Brookhaven National Laboratory. However, because of their short half-life, dilution with ambient air and the distance to the site boundary, the concentration of activity in the air is reduced to a level at which there was no detectable increase in dose equivalent at the site boundary.

Concentrations of tritium in air above background levels resulted in a maximum total-body dose at the site boundary of 0.16 millirem/y (table 3.13-2); this dose occurred about 2500 meters to the southwest of the HFBR stack. The maximum dose corresponds to a lifetime risk of fatal cancer of 2.2E-6 (table 3.13-3).

The dose commitment to the population within 80 kilometers from atmospheric emissions of tritium from BNL in 1977 was 19.2 person-rem. This would correspond to about 3.8E-3 fatal cancers per year of operations at Brookhaven National Laboratory (table 3.5-3).

Table 3.13-2. Radiation doses from atmospheric emissions of radionuclides from Brookhaven National Laboratory, 1977 (DOE77b)

Crital organ & Radionuclide	Maximum individual Site boundary (mrem)	Nearest resident (mrem)	Population[a] (person-rem)
Total body Tritium	0.16	NR	19.2

[a] Total body dose to the population within 80 kilometers.
NR Not reported.

Table 3.13-3. Individual lifetime risks and number of fatal cancers due to radioactive atmospheric emissions from Brookhaven National Laboratory, 1977

Source	Individual lifetime risks Site boundary	Average individual Region[a]	Expected fatal cancers per year of operation[b] (Fatal cancers)
BNL	2.2E-6	5.3E-8	3.8E-3

[a] The region extends to 80 kilometers.
[b] To the population within 80 kilometers.

REFERENCES

DOE77a Department of Energy, 1977, Effluent Information System (EIS) Report No. 02, Narrative Summary Data Base Master List, (EIS-02) (Computer Listing).

DOE77b Department of Energy, 1977, Effluent Information System (EIS) Report No. 51, Release Point Analysis Report for Calendar Year 1977, (EIS-51) (Computer Listing).

ERDA77 Energy Research and Development Administration 1977, Final Environmental Impact Statement, ERDA-1540, Brookhaven National Laboratory, Upton, New York.

Na78 Naidu, J.R. Ed., 1978, 1977 Environmental Monitoring Report, Brookhaven National Laboratory, BNL-50813, UC-11.

3.14 Oak Ridge Facilities

3.14.1 General Description

The three major facilities at the U.S. Department of Energy Reservation at Oak Ridge, Tennessee, are the Oak Ridge National Laboratory (ORNL), the Oak Ridge Gaseous Diffusion Plant (ORGDP), and the Y-12 Plant. Two smaller facilities located on the reservation are the Comparative Animal Research Laboratory and the Oak Ridge Associated Universities.

The Oak Ridge Reservation is located in a valley between the Cumberland and Great Smokey Mountains in eastern Tennessee and consists of approximately 15,000 hectares of government-owned land. The area is bordered to the south and west by the Tennessee Valley Authority's (TVA) Melton Hill and Watts Bar Reservoirs on the Clinch River and to the north by the city of Oak Ridge (figure 3.14-1). The surrounding area is rural with the largest population center, Knoxville, Tennessee, about 24 kilometers to the east. About 689,000 people live within 80 kilometers of the Oak Ridge National Laboratory.

3.14.2 Process Description

Oak Ridge National Laboratory

The Oak Ridge National Laboratory is a multidiscipline research laboratory; its mission is the discovery of new knowledge in all areas related to energy. Nuclear energy research facilities consist of nuclear reactors, chemical plot plants, research laboratories, radioisotope production laboratories and support facilities.

The central radioactive gas disposal facilities release tritium, iodine-131, and noble gases, krypton and xenon from radioisotope separations, reactor operations, and handling radioactive material in hot laboratories and chemistry laboratories. The gases undergo HEPA filtration at their source prior to discharge to the system. The stack is constantly monitored and sampled.

The stack servicing the High Flux Isotope Reactor and the Transuranic Processing Plant releases fission product gases resulting from the chemical separation of curium and californium and from reactor operations. Process effluent gases undergo HEPA filtration.

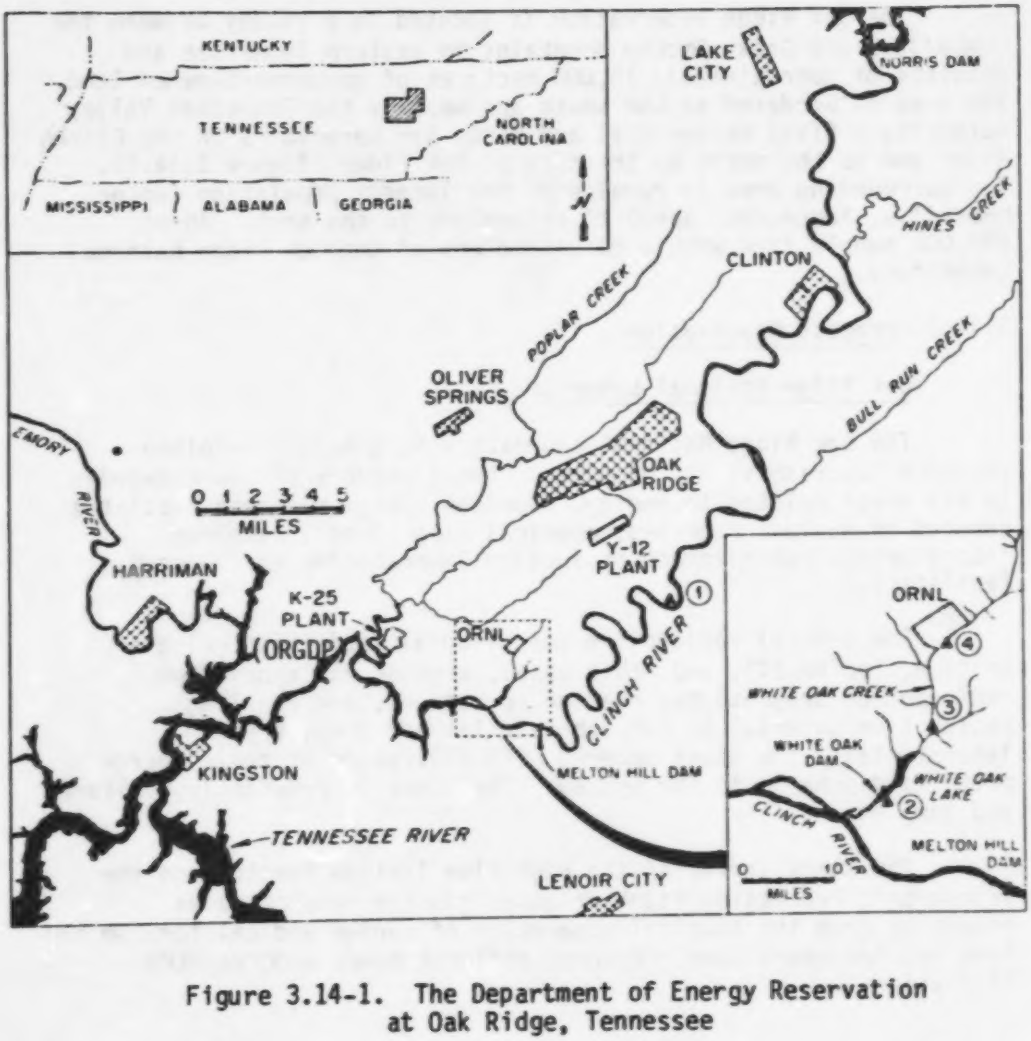
KENTUCKY
TENNESSEE
NORTH CAROLINA
MISSISSIPPI
ALABAMA
GEORGIA
N
LAKE CITY
NORRIS DAM
HINES CREEK
CLINTON
POPLAR CREEK
BULL RUN CREEK
OLIVER SPRINGS
EMORY RIVER
OAK RIDGE
0 1 2 3 4 5
MILES
HARRIMAN
Y-12 PLANT
①
K-25 PLANT
(ORGDP)
ORNL
CLINCH RIVER
KINGSTON
MELTON HILL DAM
TENNESSEE RIVER
LENOIR CITY
ORNL
④
WHITE OAK CREEK
③
WHITE OAK DAM
WHITE OAK LAKE
②
CLINCH RIVER
MELTON HILL DAM
0 10
MILES

Figure 3.14-1. The Department of Energy Reservation at Oak Ridge, Tennessee

Isotope separations and chemistry laboratory operations are the principal source of effluents. Uranium and plutonium are present in airborne effluent from the electromagnetic isotope separations facility. There are 14 exhaust points from this facility. All effluents are exhausted through one or two stages of HEPA filtration. Oil traps are also used.

A tritium target fabrication building releases small amounts of tritium from target preparation operations.

HEPA filters are used to reduce particulate activity from the transuranic research and the metal and ceramics laboratories. The effluents are monitored for alpha activity.

Oak Ridge Gaseous Diffusion Plant

The Oak Ridge Gaseous Diffusion Plant , a complex of production, research, development and support facilities, has the primary function to enrich uranium hexaflouride (UF_6) in the uranium-235 isotope.

The principal sources of release from ORGDP are the drum dryers in the decontamination facilities, which are in the uranium system, and the purging of light contaminants from the purge cascade. During 1977 the old purge cascade which used sodium fluoride and alumina traps to reduce emissions was replaced by a new purge cascade vent which has a KOH gas scrubber in the emission system.

Y-12 Plant

The Oak Ridge Y-12 Plant has four primary responsibilities: (1) production of nuclear weapons components, (2) fabrication support for weapons design, (3) support for the Oak Ridge National Laboratory, and (4) support and assistance to other government agencies. The Y-12 Plant conducts activities which include production of lithium compounds, recovery of enriched uranium from scrap material, and fabrication of uranium into finished parts and assemblies. Fabrications operations include vacuum casting, arc melting, powder compaction, rolling, forming, heat treating, machining, inspection, and testing. Many of these procedures release particulate activity into the room exhaust air. Laboratory and room air exhaust systems are equipped with filtration systems which may include prefilters, HEPA filters, or bag filters.

Oak Ridge Associated Universities

The Oak Ridge Associated Universities conducts research in areas such as biological chemistry, immunology, nuclear medicine, and radiochemistry. Radionuclides are handled in encapsulated or liquid form and the potential for producing gaseous effluents is very small.

3.14.3 Emissions of Radionuclides

The principal radioactive atmospheric emissions are tritium, krypton-85, and xenon-133 from the Oak Ridge National Laboratory. Table 3.14-1 summarizes the radioactive airborne emissions from the Oak Ridge Facilities for 1977.

Table 3.14-1. Atmospheric emissions of radionuclides
Oak Ridge Reservation, 1977 (DOE77b)

Facility	Radionuclide	Emissions (Ci/y)
Oak Ridge National Laboratory		
Central Radioactive Gas Disposal Ficilities	Tritium	2.5E+3
	Iodine-131	1.2
	Krypton-85	6.8E+3
	Xenon-133	3.3E+4
High Flux Isotope Reactor and TRU Processing Plant	Iodine-131	1.8E-1
	Krypton-85	1.8E+3
	Xenon-133	8.7E+3
Electromagnetic Isotope Separations Facility	Plutonium-239	4.0E-6
	Uranium-233	1.0E-6
Tritium Target Fabrication Building	Tritium	2.4E+1
Transuranic Research and Metal and Ceramics Laboratories	Unidentified Alpha	2.0E-8
Oak Ridge Gaseous Diffusion Plant		
Decontamination Facility	Uranium-234	5.6E-4
	Uranium-235	2.0E-5
	Uranium-236	1.0E-5
	Uranium-238	3.3E-4
Old Purge Cascade	Technetium-99	1.0E-6
	Uranium-234	4.3E-4
	Uranium-235	1.7E-5
	Uranium-236	7.4E-7
	Uranium-238	7.5E-5

Table 3.14-1. Atmospheric emissions of radionuclides Oak Ridge Reservation, 1977 (DOE77b)--continued

Facility	Radionuclide	Emissions (Ci/y)
New Purge Cascade	Technetium-99	1.0E-6
	Uranium-234	1.4E-4
	Uranium-235	5.6E-6
	Uranium-236	2.5E-7
	Uranium-238	2.5E-5
Y-12 Plant	Uranium-234	5.8E-2
Oak Ridge Associated Universities		
	Carbon-14	2.5E-4
	Tritium	1.4E-3
	Mercury-203	5.0E-5
	Iodine-131	4.6E-5

3.14.4 Health Impact Assessment of Oak Ridge Facilities

The 50-year dose commitment from 1977 atmospheric emissions to an individual occupying the nearest residence at the site boundary would result from inhalation. The maximum reported dose commitment was 4.6 millirem to the lung of an Oak Ridge resident from inhalation of uranium-234.

From milk samples taken in the immediate environs of the Oak Ridge Reservation, a maximum food-chain pathway dose commitment was reported to be 1.5 millirem to the thyroid and 5.5 millirem to the bone. These doses were based on iodine-131 and strontium-90 concentrations found in milk samples at different locations.

The cumulative total body dose to the population within 80 kilometers of the Oak Ridge facilities from 1977 atmospheric emissions was estimated to be 4.7 person-rem. This corresponds to an estimated 9.4E-4 fatal cancers per year of operation of the Oak Ridge facilities. Table 3.14-3 tabulates the individual lifetime risks associated with the doses to the critical organs.

Table 3.14-2. Radiation doses[a] from atmospheric emissions of radionuclides from Oak Ridge Facilities, 1977 (UC79)

Critical Organ and Radionuclide	Maximum individual		Population[b] (person-rem)
	Milk stations (mrem)	Nearest resident (mrem)	
Lung			
Uranium-234		4.6	NR
Bone			
Strontium-90	5.5	NR	NR
Thyroid			
Iodine-131	1.5	NR	NR
Total body			4.7

[a] Fifty-year dose commitment. Doses are not additive because they are for different locations. The nearest resident is located at the site boundary.

[b] Sum of the total body doses to exposed individuals within 80 kilometers.

NR Not reported.

Table 3.14-3 Individual lifetime risks and number of fatal cancers due to radioactive atmospheric emissions from Oak Ridge Facilities, 1977

Organ	Individual lifetime risks Maximum individual[a]	Expected fatal cancers per year of operation[b] (Fatal cancers)
Lung	1.3E-5	NR
Bone	1.2E-5	NR
Thyroid	1.1E-7	NR
Total body	NR	9.4E-4

[a] Risks are not additive since they correspond to maximum individual doses that occur at different locations.

[b] To the population within 80 kilometers.

NR Not reported.

REFERENCES

DOE77a Department of Energy, 1977, Effluent Information System Report Nos. 02 and 05, Narrative Summary Data Base MAster LIst and Narrative Summary Data Base Master list Updated Report for 1977, (EIS 02 and EIS 05), (Computer Report).

DOE77b Department of Energy, 1977, Effluent Information System Report No. 51. Release Point Analysis Report for Calendar Year 1977, (EIS 51), (Computer Report).

UC79 Union Carbide Corporation-Nuclear Division, 1978, Environmental Monitoring Report United States Department of Energy Oak Ridge Facilities Calendar Year 1977, Y/UB-8.

3.15 Portsmouth Gaseous Diffusion Plant

3.15.1 General Description

The Portsmouth Gaseous Diffusion Plant has been operated by the Goodyear Atomic Corporation for the Department of Energy since 1954. The principal process at the plant is the enrichment of uranium in the uranium-235 isotope by separation of uranium isotopes through gaseous diffusion.

The Portsmouth Plant, located on 1619 hectares of Federally-owned land in Pike County, Ohio, is situated in the south-central part of the State about 1.6 kilometers east of the Scotia River Valley and approximately 32 kilometers north of Portsmouth, Ohio. The area around the plant is predominately rural with marginal farm land and densely forested hills. Pike County is sparsely populated; less than 600,000 people live within 80 kilometers of the plant site.

3.15.2 Process Description

The Portsmouth Gaseous Diffusion Plant cascade is the only facility in the United States with the capability of producing a very highly enriched uranium product (VHE = 97.65% uranium-235). The cascade consists of 4020 isotopic stages that produce enriched uranium-235 and 60 purge stages that separate and purge the light gas contaminants that leak into the system. Three imposing buildings (X-326, X-330, X-333) house the gaseous diffusion process equipment. The principal points of release of radioactivity from the process buildings are from the top purge vent in building X-326, and the cold-recovery system vent in building X-330. Two other support facilities are also sources of emissions at the site.

The purge facility separates the light contaminants (air, N_2, HF and coolant) from the process gas flow. The light contaminant gases are passed through alumina traps to an air-jet exhauster and discharged to the atmosphere from a roof-top vent. Purge gases collected from locations throughout the plant are passed through refrigerated cold traps which are a part of the X-300 cold-recovery system. These cold traps freeze out UF_6 from the gases which are then passed through NaF traps for removal of remaining traces of UF_6 before being released to the atmosphere through a vent on a building roof by means of air-jet exhausters.

Building X-705, the decontamination facility, is centrally located with respect to the three process buildings. This facility provides space, special handling equipment, and fixtures for disassembly, decontamination and radiation monitoring of cascade components that are removed for repair. The principal source of airborne radioactive emissions is the process of converting uranium oxide (U_3O_8) to uranium hexafluoride (UF_6) from uranium recovered from decontamination solutions. The UF_6 is then reintroduced as feed to the cascade. The UF_6 in the off-gas stream is removed by a cold trap. Off-gases from the cold trap are passed through sodium fluoride traps . Particulates in the gas stream are removed prior to the cold trap by a sintered metal filter, and volatile impurities are adsorbed on a magnesium flouride trap.

The remaining facility at the site that is an important source of emissions is building X-744G which is used for receiving, sampling, storaging, transferring and shipping of licensed and DOE-owned uranium materials. Sampling and transferring operations are sources of releases of UF_6. The gases are passed through cold traps and alumina traps before being vented to the atmosphere.

3.15.3 Emissions of Radionuclides

Table 3.15-1 summarizes the radioactive airborne emissions from the four release points at the Portsmouth Gaseous Diffusion Plant in 1977.

Table 3.15-1. Atmospheric emissions of radionuclides from Portsmouth Gaseous Diffusion Plant, 1977 (DOE77b)

Facility	Radionuclide	Emissions (Ci/y)
Building X-326	Protactinium-234m	4.0E-2
Top Purge	Technetium-99	3.2
Cascade Vent	Thorium-234	4.0E-2
	Uranium-234	9.5E-2
	Uranium-235	3.3E-3
	Uranium-236	1.1E-4
	Uranium-238	5.0E-4
Building X-330	Protactinium-234m	5.9E-3
Cold Recovery	Technetium-99	1.3
System Vent	Thorium-234	5.9E-3
	Uranium-234	2.8E-3
	Uranium-235	1.7E-4
	Uranium-236	8.3E-7
	Uranium-238	4.1E-3
Building X-705	Uranium-234	9.4E-4
Building	Uranium-235	6.0E-5
Waste Gas Vent	Uranium-236	3.1E-6
	Uranium-238	1.4E-5
Building X-744G	Protactinium-234m	1.6E-5
Oxide	Technetium-99	2.0E-3
Sampling Facility	Thorium-234	1.6E-5
	Uranium-234	1.7E-4
	Uranium-235	1.1E-5
	Uranium-236	4.5E-7
	Uranium-238	1.1E-6

3.15.4 Health Impact Assessment of Portsmouth Gaseous Diffusion Plant

Table 3.15-2 summarizes the 50-year dose commitment from radioactive airborne emissions from the Portsmouth Gaseous Diffusion Plant. The individual doses reported were based on the maximum average alpha and beta-gamma radioactivity concentrations obtained from air sampler data. The maximum dose at the eastern site boundary and the maximum off-site residence dose were considered the same since a few families reside at the property boundary. The nearest community for which these doses apply is the town of Piketon, about 8 kilometers north of the plant.

Table 3.15-3 summarizes the estimated individual lifetime risk of fatal cancer and the number of fatal cancers resulting from plant operations.

Table 3.15-2. Radiation doses[a] from atmospheric emissions of radionuclides from Portsmouth Gaseous Diffusion Plant, 1977 (An78)

Organ	Maximum individual Site boundary (mrem)	Average individual in the nearest community[b] (mrem)	Population[c] (person-rem)
Lung	1.12	3.9E-2	NR
Bone	1.23	4.9E-2	NR
Kidney	1.06	4.1E-2	NR
G.I. tract	2.16	8.5E-2	NR
Total body	0.21	8.0E-3	0.19

[a] The maximum individual and average individual doses are based on measured airborne radioactivity concentrations. Doses are 50-year dose commitments.

[b] Piketon is located approximately 8 kilometers from the plant.

[c] Total body dose to the population within 80 kilometers.

NR Not reported.

Table 3.15-3. Individual lifetime risks and number of fatal cancers due to radioactive atmospheric emissions from Portsmouth Gaseous Diffusion Plant, 1977

	Individual lifetime risks			
Organ	Maximum individual	Nearest community[b]	Average individual Region[a]	Expected fatal cancers per year of operation[c] (Fatal cancers)
Lung	3.1E-6	1.1E-7	NR	NR
Bone	2.6E-6	1.0E-7	NR	NR
Kidney	7.4E-7	2.9E-8	NR	NR
G.I. tract	3.0E-6	1.2E-7	NR	NR
Other soft tissue	7.4E-7	2.8E-8	NR	NR
Total	1.0E-5	3.9E-7	4.4E-9	3.8E-5

[a] The region extends to 80 kilometers.

[b] Piketon.

[c] To the population within 80 kilometers.

NR Not reported.

REFERENCES

An78 Anderson, Robert E., Bonnie J. Rumble, and Edgar R. Warner, 1978, Portsmouth Gaseous Diffusion Plant Environmental Monitoring Report for Calendar Year 1977, GAT-955, Piketon, Ohio

DOE77a Department of Energy, 1977, Effluent Information System Report No. 02, Narrative Summary Data Base Master List, EIS 02, (Computer Report).

DOE77b Department of Energy, 1977, Effluent Information System Report No. 51, Release Point Analysis Report for Calendar Year 1977, EIS 51, (Computer Listing)

ERDA77 Energy Research and Development Administration, 1977 Final Environmental Impact Statement, Portsmouth Gaseous Diffusion Plant Site, Piketon, Ohio, ERDA-1555, Washington, D.C.

3.16 Paducah Gaseous Diffusion Plant

3.16.1 General Description

The Paducah Gaseous Diffusion Plant (PGDP) is a uranium enrichment cascade plant with a uranium hexafluoride (UF_6) manufacturing plant and various other support facilities.

The Paducah Gaseous Diffusion Plant, located in McCracken County, Kentucky, is 19 kilometers west of Paducah, Kentucky, and about 6 kilometers south of the Ohio River. The plant is within a 300-hectare fenced area. The site is encircled by a buffer zone of Government-owned land and beyond that, a wildlife management area. The nearest incorporated town is Metropolis, Illinois, about 8 kilometers to the northeast.

3.16.2 Process Description

The primary plant is the Diffusion Cascade Plant. The principal process is the separation of uranium isotopes through gaseous diffusion. Uranium hexafluoride (UF_6) gas is fed into the system and pumped through up to 1812 stages in the enrichment cascade. The product UF_6 is enriched in the uranium-235 isotope. The "tails" are withdrawn as UF_6 depleted in uranium-235.

All the stages in the enrichment cascade are contained within five buildings. The prime source of emissions is from the purge cascade which is used for removal of light contaminants from the process stream. These contaminants, which consist of isotopes of uranium and technetium-99, are released from the diffusion cascade building stack which is sampled regularly.

The manufacturing building or Feed Plant uses hydrogen, anhydrous hydrogen fluoride (HF) and uranium oxide (UO_3) to produce the UF_6 that is fed into the diffusion cascade. Gaseous emissions, from fluorination operations of UF_4 to UF_6, are passed through a series of waste treatment systems that include cold traps, fluid bed absorbers and sintered metal filters. HEPA and bag filters are also used to treat other emissions from the Feed Plant.

The Uranium Recovery and Chemical Processing Facility conducts operations that involve pulverizing and screening of uranium salts. Here bag filters are used to reduce airborne emissions.

At the Metals Plant, depleted UF_6 from the Cascade is reacted with HF to convert it to UF_4 which is more easily stored.

Table 3.16-1. Atmospheric emissions of radionuclides from Paducah Gaseous Diffusion Plant, 1977 (DOE77b)

Facility and Radionuclide	Emissions (Ci/y)
Diffusion Cascade	
Technetium-99	6.8E-2
Uranium-234	9.0E-3
Uranium-235	3.8E-4
Uranium-236	5.0E-6
Uranium-238	3.1E-3
Feed Plant	
Uranium-234	1.5E-1
Uranium-235	6.7E-3
Uranium-236	2.3E-4
Uranium-238	1.5E-1
Uranium Recovery and Chemical Processing	
Uranium-234	1.6E-3
Uranium-235	1.4E-4
Uranium-236	5.1E-5
Uranium-238	1.1E-2
Metals Plant	
Uranium-234	7.1E-4
Uranium-235	6.0E-5
Uranium-236	2.3E-5
Uranium-238	4.8E-3
Warehouse	
Uranium-234	3.5E-2
Uranium-235	1.6E-3
Uranium-236	5.2E-5
Uranium-238	3.5E-2

Bag filters are used for these operations to reduce emissions. Also within the same facility UF_4 can be reduced to metal by reaction with magnesium. However, there were no such operations in 1977.

Ash which is collected at the bottom of the flame tower in the process of fluorinating uranium oxides at the Feed Plant is taken to a warehouse for storage. The ash, containing isotopes of uranium, may become airborne and be released to the atmosphere. There are no effluent control and exhaust air systems; however, emissions are continuously sampled.

3.16.3 Emissions of Radionuclides

The radioactive airborne emissions from the Paducah Gaseous Diffusion Plant consisted of four isotopes of uranium and technetium-99. Table 3.16-1 summarizes the emissions for 1977.

3.16.4 Health Impact Assessment of Paducah Gaseous Diffusion Plant

Table 3.16-2 summarizes the 50-year dose commitment to the bone and lung from inhalation of uranium emitted from Paducah Gaseous Diffusion Plant in 1977. The maximum site boundary dose was found to occur to the north of the site. The nearest resident is approximately 2.2 kilometers east of the plant boundary. The potential dose to the nearest resident would correspond to a lifetime risk of fatal cancer of 2.8E-6. No population dose was reported.

Table 3.16-2. Radiation doses[a] from atmospheric emissions of radioactive materials at Paducah Gaseous Diffusion Plant, 1977 (UC78)

Organ	Maximum site boundary dose[b] (mrem)	Nearest resident dose[c] (mrem)
Lung	6.3	0.9
Bone	0.9	0.1

[a] Fifty-year dose commitment.

[b] Location--north of site.

[c] Approximately 2.2 kilometers east of the plant boundary.

Table 3.16-3. Individual lifetime risks and number of fatal cancers resulting from atmospheric emissions of radionuclides at Paducah Gaseous Diffusion Plant, 1977

Source	Individual lifetime risk Site boundary	Nearest resident	Expected fatal cancers per year of operation (Fatal cancers)
Paducah Gaseous Diffusion Plant	2.0E-4	2.8E-6	NR

NR Not reported.

REFERENCES

DOE77a Department of Energy, 1977, Effluent Information System Report No. 2, Narrative Summary Data Base Master List, (EIS 02), (Computer Listing).

DOE77b Department of Energy, 1977, Effluent Information System Report No. 51, Release Point Analysis Report for Calendar Year 1977, (EIS 51), (Computer Listing).

UC78 Union Carbide Corporation, 1978, Environmental Monitoring Report, United States Department of Energy, Paducah Gaseous Diffusion Plant, Calendar Year 1977, Y/UB-9, Office of Health, Safety, and Environmental Protection, P. O. Box Y, Oak Ridge, Tenn. 37830.

3.17 Ames Laboratory

3.17.1 General Description

Ames Laboratory is operated by Iowa State University for the Department of Energy. The principal facility is the Ames Laboratory Research Reactor (ALRR) along with its associated radioactive waste disposal/decontamination facility. Other facilities on campus include offices, laboratories, warehouses and service buildings.

The reactor facility is located about 2.4 kilometers northwest of the campus and about 4.8 kilometers west and north of Ames on approximately 16.2 hectares of land. Ames is located in an agricultural area of central Iowa. The population of Ames is 40,000, half of which is comprised of students. The total population within 80 kilometers is about 590,500.

3.17.2 Process Description

The ALRR is fueled with uranium-235 and uses heavy water as a coolant and moderator which is circulated in a closed system. The principal uses of the 5 MW reactor, which operates on a continuous schedule, are to produce fission products for research, to irradiate crystals for neutron diffraction studies, to irradiate metals for the study of radiation effects, to produce isotopes for tracer studies, and to activate various elements for analytical studies.

Operations within the disposal/decontamination facility include solid waste compaction, liquid storage in tanks with filtration and ion exchange processing, evaporation and distilation, and packaging of solid waste for shipment to a radioactive disposal site.

Airborne emissions from reactor operations are filtered and monitored before release.

3.17.3 Emissions of Radionuclides

The principal emissions from Ames Laboratory Research Reactor are argon-41 and tritium. Table 3.17-1 summarizes the emissions from Ames Laboratory for 1977.

Table 3.17-1. Atmospheric emissions of radionuclides, Ames Laboratory, 1977 (DOE77)

Facility and Radionuclide	Emissions (Ci/y)
Ames Laboratory Research Reactor (ALRR)	
Argon-41	1.3E+4
Beryllium-7	9.0E-5
Cerium-141	2.9E-6
Cerium-144	1.5E-5
Cesium-137	9.1E-7
Tritium	1.0E+3
Iodine-131	6.4E-7
Niobium-95	4.2E-5
Ruthenium-103	2.1E-5
Ruthenium-106	3.2E-6
Unidentified Alpha	8.3E-6
Unidentified Beta & Gamma	2.8E-5
Zirconium-95	1.3E-5
Radioactive Waste Disposal Decontamination Facility	
Tritium	9.1E-2

3.17.4 Health Impact Assessment of Ames Laboratory

The doses reported for Ames Laboratory were estimated by applying principles of meteorological diffusion. Table 3.17.2 summarizes the dose to an average individual at the site boundary (about 213 meters from the reactor) and to the population residing within 80 kilometers.

An individual exposed to the total body dose reported in table 3.17.2 would have an estimated lifetime risk of fatal cancer of 7.7E-5. An estimated 4.9E-2 fatal cancers are estimated per year of operation of ALRR to the population residing within 80 kilometers.

Table 3.17-2. Annual radiation doses from atmospheric emissions of radioanuclides from Ames Laboratory, 1977 (Vo78)

Radionuclide	Average individual Site boundary (mrem/y)	Population[a] (person-rem/y)
Argon-41	5.4	4.6
Tritium	0.1	240
Total	5.5	245

[a]Total body dose to the population within 80 kilometers.

Table 3.17-3. Individual lifetime risks and number of fatal cancers due to radioactive atmospheric emissions from Ames Laboratory, 1977

Organ	Individual lifetime risks		Expected fatal cancers per year of operation (Fatal cancers)
	Site boundary	Average individual Region[a]	
Total body	7.7E-5	5.6E-6	4.9E-2

[a]The region extends to 80 kilometers.
[b]To the population within 80 kilometers.

REFERENCES

DOE77 Department of Energy, 1977, Effluent Information System Report No. 51, Release Point Analysis Report for Calendar Year 1977, (EIS 51), (Computer Report).

Vo78 Voss M. D., 1978, Environmental Monitoring Report at Ames Laboratory, Calendar Year 1977, 15-4365, Ames, Iowa.

3.18 Atomics International

3.18.1 General Description

Atomics International , a Division of Rockwell International Corporation, is engaged in research and development of atomic energy. Efforts include design, development fabrication and testing of systems and components for central station power plants, fabrication of nuclear fuel for test reactors, and decontamination and disposition of facilities.

Atomics International operates two sites in California. At the Canoga Park site--which is licensed by NRC and the State of California--programs utilize uranium fuel production facilities, analytical chemistry laboratories, and a cobalt-60 irradiation facility. The other site, the Santa Susana Field Laboratories (SSFL), is approximately 46 kilometers northwest of Los Angeles, California, on a 117-hectare site. Both DOE and Rockwell share this site. The SSFL facilities, which are licensed by the NRC and the State of California include a hot laboratory, a nuclear materials development facility, a neutron radiography facility containing a research reactor, and several X-radiography inspection facilities. DOE contract activities are conducted within a 33-hectare area at the SSFL site.

3.18.2 Process Description

DOE operations at SSFL which may release radioactive materials into the atmosphere are conducted at the Radioactive Material Disposal Facility (RMDF). The two buildings (021-022) that comprise this facility are used for processing wastes generated by a program for the decontamination and disposition of DOE facilities. Liquid and dry radioactive wastes are processed, packaged and temporarily stored for further disposal. Nuclear fuel material, handled in encapsulated or encapsulated forms, contains the isotopes uranium-234, uranium-235, uranium-236, uranium-238, cesium-137, strontium-90 and promethium-147. Airborne emissions from this facility are reduced by exhausting air through a HEPA filtration system.

NRC and California State Licensed Activities include hot cell operations conducted in Building 020. Here irradiated nuclear fuels and reactor components are examined. Radioactive material handled in unencapsulated form at this facility includes the following radionuclides: thorium-232, uranium-233, uranium-234, uranium-235, uranium-236, and uranium-238 as constituents in the various fuel materials; and cesium-137, strontium-90, krypton-85 and promethium-147 as mixed fission products. Airborne emissions may contain radioactive particulates and gases, depending on the operations being performed.

Operations at Building 055, that are licensed by the NRC and the State of California, involve the fabrication of plutonium and plutonium-uranium fuel pins. The fuel may consist of depleted and enriched uranium and plutonium materials.

3.18.3 Emissions of Radionuclides

Table 3.18-1 summarizes the radioactive emissions from the Santa Susana Field Laboratories. The DOE emissions were less than eight percent of the total emissions reported from all the facilities at both Atomics International sites.

Table 3.18-1. Atmospheric emissions of radionuclides from the Santa Susana Field Laboratories, 1977 (Mo78)

Location/Radionuclide	Emissions (Ci/y)
Buildings 021-022	
Unidentified Alpha	<1.1E-7
Unidentified Beta	<3.0E-6
Building 020	
Unidentified Alpha	<1.0E-7
Unidentified Beta	1.3E-5
Building 055	
Unidentified Alpha	<1.5E-7

3.18.4 Health Impact Assessment of Atomics International

Dose calculations were not reported for Atomics International facilities. Moore reported (Mo78) that the gaseous effluent release from the facilities is less radioactive than the ambient air.

REFERENCES

DOE77a Department of Energy, 1977, Effluent Information System Report No. 02, Narrative Summary Data Base Master List, (EIS 02). (Computer Report).

DOE77b Department of Energy, 1977, Effluent Information System Report No. 51, Release Point Analysis Report for Calendar Year 1977, (EIS 51), (Computer Report).

Mo78, More J. D., 1978, Atomics International Environmental Monitoring and Facility Effluent Annual Report 1977, Canoga Park, California.

3.19 Battelle Columbus Laboratory

3.19.1 General Description

Battelle Columbus Laboratory (BCL) conducts various NRC-licensed activities as well as activities under Department of Energy contracts.

BCL operates two complexes in the Columbus, Ohio area. The first site is the King Avenue site, which consists of four hectares of land near a residential area in Columbus. The Ohio State University intramural sports practice field borders the site to the north.

The second site is the Nuclear Sciences Area of the West Jefferson site, which is located about 27 kilometers west of the King Avenue laboratories. This site occupies about four hectares on a 405-hectare tract of land. There are approximately 623,400 people living within 80 kilometers of the laboratory.

3.19.2 Process Description

The King Avenue site has a uranium-235 processing facility located within Building 3. This building also houses the melting facility and the powder metallurgy laboratory. The uranium processing facility manages all transactions involving nuclear material at the King Avenue site. However, handling of contract and licensed material was very limited in 1977 and there have been no reported airborne emissions since 1975.

At the West Jefferson site activities at the Nuclear Sciences Area include: JN-1 hot cell operations where irradiated reactor fuel elements are studied; JN-4 plutonium laboratory work, where research is conducted on uranium-235/plutonium-239 nitride reactor fuel; and materials accountability and storage operations, conducted at the JN-2 vault.

Airborne radioactive emissions at the Battelle Columbus Laboratory are first filtered at the points of operations i.e., glove boxes, hoods, test cells and then passed through one or two stages of HEPA filters before release. The hot cell facility is equipped with a charcoal bed so radioactive gases can be routed through it when necessary.

3.19.3 Emissions of Radionuclides

Table 3.19-1 summarizes the radioactive emissions from the West Jefferson site in 1977. There were no reported emissions from DOE contract activities from the King Avenue site.

3.19.4 Health Impact Assessment of Battelle Columbus Laboratory

Table 3.19.2 summarizes the estimated annual radiation doses from radioactive emissions from the West Jefferson site of Battelle Columbus Laboratory. The maximum dose at the site boundary is considered coincident with the downwind position from the facility where the highest annual concentrations of radionuclides will occur. This point, for uncontrolled exposure, is outside the security fence but still on BCL property. The table also shows the maximum dose estimate for an individual in the nearest population group, a distance of two kilometers.

Table 3.19-3 estimates the individual lifetime risks and number of fatal cancers in the population resulting from these doses. For each year of operation of BCL there would be an estimated 2.1E-7 fatal cancers in the population within 80 kilometers.

Table 3.19-1. Radioactive airborne emissions from Battelle Columbus Laboratory, 1977 (Ev78)

Facility	Radionuclide	Emissions (Ci/y)
JN-1 Hot Cell	Barium-133	6.0E-9
	Cesium-134	1.3E-7
	Cesium-137	1.3E-6
	Chromium-51	1.3E-8
	Cobalt-57	3.3E-9
	Cobalt-60	1.4E-4
	Europium-152	7.9E-7
	Europium-154	4.5E-7
	Mercury-203	4.2E-7
	Iodine-131	1.5E-7
	Iridium-192	4.0E-9
	Krypton-85	5.2E-6
	Lanthanum-140	2.9E-5
	Manganese-54	1.6E-7
	Niobium-95	4.0E-8
	Ruthenium-97	1.4E-7
	Antimony-12	2.6E-7
	Antimony-125	3.5E-5
	Tin-113	6.0E-8
	Terbium-160	3.2E-7
	Unidentified Alpha	1.8E-7
	Unidentified Beta	4.8E-6
	Uranium-235	2.0E-7
	Xenon-133m	2.9E-7
	Zinc-65	9.0E-8
	Zirconium-95	2.1E-7
JN-4 Plutonium Laboratory	Plutonium-239	7.2E-8
JN-2 Vault	Plutonium-239	7.0E-9

Table 3.19-2. Annual radiation doses from atmospheric emissions of radioactive materials from Battelle Columbus Laboratory, 1977 (Ev78)

Critical organ	Maximum individual Site boundary (mrem/y)	Nearest resident (mrem/y)	Population[a] (person-rem/y)
Lung	1.1E-2	6.5E-06	1.6E-3
Bone	7.0E-2	4.1E-05	4.8E-3
Kidney	1.2E-3	6.8E-07	7.9E-5
Thyroid	5.5E-5	3.2E-08	3.8E-6
G. I. tract	3.4E-4	2.0E-07	2.3E-5
Skin	6.4E-7	3.8E-10	4.4E-8
Other soft tissue	1.3E-7	7.2E-11	8.4E-9

[a] Total body dose to the population within 80 kilometers.

Table 3.19-3. Individual lifetime risks and number of fatal cancers due to radioactive atmospheric emissions from Battelle Columbus Laboratory, 1977

Critical organ	Individual lifetime risks Site boundary	Nearest resident	Expected fatal cancers per year of operation (Fatal cancers)[a]
Lung	3.1E-08	1.8E-11	6.4E-08
Bone	1.5E-07	8.6E-11	1.4E-07
Kidney	8.4E-10	4.8E-13	7.9E-10
Thyroid	3.9E-12	2.2E-14	3.8E-12
G. I. tract	4.8E-10	2.8E-13	4.6E-10
Skin	2.2E-12	1.3E-15	2.2E-12
Other soft tissue	4.6E-13	2.5E-16	4.2E-13
Total	1.8E-07	1.0E-10	2.1E-07

[a] To the population within 80 kilometers.

REFERENCES

Ev78 Evans R.G., 1978, Environmental Report for Calendar Year 1977 on Radiological and Non-Radiological Parameters to United States Department of Energy Chicago Operations Office, Battelle Columbus Laboratories, Columbus, Ohio.

3.20 Bettis Atomic Power Laboratory

3.20.1 General Description

The Bettis Atomic Power Laboratory (BAPL) is operated by the Westinghouse Electric Corporation for the Naval Reactors Division of the Department of Energy. Operations involve design and development of naval nuclear power reactors.

Bettis Atomic Power Laboratory occupies about 81 hectares of land in West Mifflin, Pennsylvania, about 13 kilometers southwest of Pittsburgh. The total population within 80 kilometers of the site is about 3.3 million.

3.20.2 Process Description

Fuel development efforts and hot cell chemistry work are the principal operations contributing to radioactive airborne emissions from BAPL. HEPA filters are used to reduce particulates in airborne effluents, and charcoal filters are used where necessary to reduce gaseous emissions. Air exhausted from the laboratory's fume hoods is treated by filtration or wet scrubbers to minimize emissions. Exhaust stacks which discharge air from areas where radioactive materials are handled are continiuosly monitored.

3.20.3 Emissions of Radionuclides

Table 3.20-1 summarizes the radioactive airborne emissions from Bettis Atomic Power Laboratory for 1977.

3.20.4 Health Impact Assessment of Bettis Atomic Power Laboratory

Table 3.20-2 summarizes the annual doses from airborne emissions from Bettis Atomic Power Laboratory in 1977. Bone was the critical organ for particulate activity and the lung was the critical organ for gaseous activity.

Table 3.20-3 estimates the individual lifetime risks of fatal cancer to a hypothetical individual at the site boundary and to the average individual in the region within 80 kilometers. The table also estimates the number of fatal cancers that would occur per year in the population within 80 kilometers for each year of operation of Bettis Atomic Power Laboratory.

Table 3.20-1. Atmospheric emissions of radioanuclides from Bettis Atomic Power Laboratory, 1977 (DOE77b)

Facility and Radionuclide	Emissions (Ci/y)
Main Laboratory and NE Area	
Tritium	4.8E-5
Iodine-131	1.1E-4
Krypton-85	8.2E-1
Antimony-125	1.4E-4
Unidentified Alpha	2.2E-5
Unidentified Beta and Gamma	2.9E-4

Table 3.20-2. Annual radiation doses from atmospheric emissions of radionuclides from Bettis Atomic Power Laboratory, 1977 (We78)

Crital organ	Maximum individual Site boundary (mrem/y)	Average individual Region (mrem/y)	Population[a] (person-rem/y)
Bone	<1.0	3.7E-3	12.3
Lung	<1.0	NR	NR
Total body	<0.1	6.1E-5	0.2

[a] To the population within 80 kilometers.
NR Not reported.

Table 3.20-3. Individual lifetime risks and number of fatal cancers due to radioactive atmospheric emissions from Bettis Atomic Power Laboratory, 1977

Critical Organ	Individual lifetime risks Maximum individual	 Average individual Region[a]	Expected fatal cancers per year of operation (Fatal cancers)[b]
Bone	2.1E-6	7.8E-09	3.7E-4
Lung	2.8E-6	NR	NR
Total body	1.4E-6	8.5E-10	4.0E-5

[a] The region extends to 80 kilometers.

[b] To the population within 80 kilometers.

NR Not reported.

REFERENCES

DOE77a Department of Energy, 1977, Effluent Information System Report No. 02, Narrative Summary Data Base List, EIS02. (Computer Report).

DOE77b Department of Energy, 1977, Effluent Information System Report No. 51, Release Point Analysis Report for Calendar Year 1977, EIS 51, (Computer Report).

WE78 Westinghouse Electric Corporation, 1978, Effluent and Environmental Monitoring Report for Calendar Year 1977, WAPD-RSC(HE)-460, Bettis Atomic Power Laboratory, Pittsburgh, PA.

3.21 Feed Materials Production Center

3.21.1 General Description

The Feed Materials Production Center (FMPC) produces purified uranium metal and compounds for use at other DOE sites.

The FMPC is located on a 425-hectare site about 32 kilometers northwest of Cincinnati, Ohio. The area east of the site is in the Miami River Flood plain which is primarily used for farming; the area downriver is sparsely populated with small and scattered industries. The total population within 80 kilometers of the FMPC site is about 2.5 million.

3.21.2 Process Description

Uranium production may begin with ore concentrates, recycled uranium from spent fuel, or with various uranium compounds. Impure starting material is dissolved in nitric acid and the uranium is extracted into an organic liquid and then back-extracted into dilute nitric acid to yield a solution of uranyl nitrate.

Evaporation and heating convert the nitrate solution to uranium trioxide (UO_3) powder. This compound is reduced to uranium dioxide (UO_2) with hydrogen and then converted to uranium tetrafluoride (UF_4) by reaction with anhydrous hydrogen fluoride. Uranium metal is produced by reacting UF_4 and magnesium metal in a refractory-lined reduction vessel. This primary uranium metal is then remelted with scrap uranium metal to yield a purified uranium ingot which is extruded to form rods or tubes. Sections are then cut and machined to final dimensions. These machined cores are shipped to other DOE sites for canning and final assembly into reactor fuel elements.

Periodically, small amounts of thorium are processed at the Center. Thorium production steps, in general, are similar to those followed in uranium production. Final products may be purified thorium nitrate solution, solid thorium compounds, or metal (Bo78).

The eight buildings or plants at the FMPC which carry out the operations just described are equipped with cloth type bag filters to reduced atmospheric emissions.

Table 3.21-1. Atmospheric emissions of radionuclides, Feed Materials Production Center, 1977 (DOE77a, DOE77b)

Facility	Process or Operation	Emissions (Ci/y)
Plant 1	Material Sampling and Grinding	-
Plant 2	Dumping Dry Feeds and Feeding Digest Tanks	2.0E-3
Plant 4	UF4 Production and Repackaging	4.0E-3
Plant 5	Metal Production and Slag Grinding	1.9E-2
Plant 6	Machining of Uranium Metal	-
Plant 8	Dumping, Milling,	1.7E-3
Plant 9	Production of Uranium Metal Remelting and Machining	-
Pilot Plant	Production of Thorium and Uranium Compounds and Metal	3.3E-3
	Total	3.0E-2

3.21.3 Emissions of Radionuclides

Table 3.21.1 summarizes the emissions of radioactivity into the atmosphere in 1977 (DOE77b) from the FMPC and describes the operations (DOE77a) associated with the emissions. The uranium-235 content of the uranium handled at the Feed Materials Production Center may he depleted, normal, or slightly enriched. However, the average content is close to that of natural uranium.

3.21.4 Health Impact Assessment of the Feed Materials Production Center

In 1977 the highest average concentration of airborne uranium occurred at the air sampling station at the eastern boundary of the site. Table 3.21-2 summarizes the doses resulting from 1977 emissions. The maximum dose to the hypothetical individual at the site boundary is estimated to be 4.8 mrem to the lung, while the nearest resident who lives near the sampling station is estimated to receive a lung dose of 2.8 mrem. The airborne uranium would result in a 50-year whole body dose commitment of 1.4 person-rem to the population within 80 kilometers.

Table 3.21-3 summarizes the individual lifetime risks and number of fatal cancers resulting from these doses.

The Town of Ross, the nearest community to the Feed Materials Production Center, is approximately 4 kilometers to the northeast of the production area. Table 3.21-4 summarizes the doses and lifetime risks of cancer to an individual living in that community.

Table 3.21-2. Radiation doses[a] from atmospheric emissions of radionuclides from the Feed Materials Production Center, 1977 (Bo78)

Crital organ	Maximum individual Site boundary (mrem)	Maximum individual Nearest resident (mrem)	Population[b] (person-rem)
Lung	4.8	2.8	NR
Total body	NR	NR	1.4

[a] Fifty-year dose commitment.
[b] Total body dose to the population within 80 kilometers.
NR Not reported.

Table 3.21-3. Individual lifetime risks and number of fatal cancers due to radioactive atmospheric emissions from the Feed Materials Production Center, 1977

Critical organ	Individual lifetime risks Site boundary	 Nearest resident	Expected fatal cancers per year of operation (Fatal cancers)[a]
Lung	1.3E-5	7.8E-6	NR
Total body	NR	NR	2.8E-4

[a] To the population within 80 kilometers.
NR Not reported.

Table 3.21-4. Radiation doses[a] and individual lifetime risks of fatal cancers to residents of the Town of Ross due to radioactive atmospheric emissions from the Feed Materials Production Center, 1977

Organ	Average individual (mrem)	Individual lifetime risks of fatal cancer to a resident of Ross
Lung	0.65	1.8E-6
Bone	0.22	4.6E-7
Kidney	0.36	2.5E-7
Other soft tissue	0.03	1.1E-7
Total		2.6E-6

[a] Fifty-year dose commitment.

REFERENCES

Bo78 Boboak M. W., K. N. Ross, and D. A. Fuchs, 1978, Feed Materials Production Center Environmental Monitoring Annual Report for 1977, NLCO-1151, Cincinnati, Ohio.

DOE77a Department of Energy, 1972, Effluent Information System Report Report No. 02, Narrative Summary Data Base Master List, EIS 02, (Computer Report).

DOE77b Department of Energy, 1977, Effluent Information System Report No. 51, Release Point Analysis Report for Calendar Year 1977, EIS 51, (Computer Report).

3.22 Knolls Atomic Power Laboratory

3.22.1 General Description

The Knolls Atomic Power Laboratory (KAPL) consists of three sites: the Knolls Site, the Kesselring Site, and the Windsor Site. At the Knolls Site, the principal function is the development of nuclear power plants, while at the Kesselring and Windsor sites the principal function is training personnel in the operation of nuclear reactors.

The Knolls Site is situated on 69 hectares of land along the Mohawk River, about eight kilometers east of Schenectady, N. Y., in a relatively low population, residential area. About 1.23 million people live within 80 kilometers of Knolls Site.

The Kesselring Site--a 1579-hectare site--is located near West Milton, NY., approximately 24.4 kilometers north of Schenectady. The surrounding area is rural and sparsely populated; about 1.08 million people live within 80 kilometers.

The Windsor Site consists of only 4 hectares of land near Windsor, Connecticut, about eight kilometers north of the city of Hartford. The area is a rural farming and industrial region along the Farmington River. Approximately 3.1 million people live within 80 kilometers.

3.22.2 Process Description

The principal operations which are potential sources of emissions of radioactive materials into the atmosphere at the Knolls Site include: hot cell operations, the use of critical assemblies, the handling of radioactive materials in chemistry and physics laboratories, and the operation of a uranium metallurgy laboratory. These operations, along with any other activities involving radioactive materials, are serviced by controlled exhaust systems which discharge through elevated stacks. Exhaust air is passed through HEPA and carbon filters and continuously sampled prior to release.

The Kesselring site has three pressurized water nuclear reactor plants and associated support facilities. Particulate and gaseous activity contained in the primary coolant may become airborne from reactor coolant discharges, sampling operations and during laboratory operations. Exhaust air from these operations is passed through HEPA filters, monitored, and released from elevated stacks.

The Windsor Site contains one pressurized water nuclear reactor plant used for training. As at the Kesselring Site, the exhaust air from operations which have the potential for release of airborne radioactivity is passed through HEPA filters, then monitored, and released through elevated stacks.

3.22.3 Emissions of Radionuclides

Table 3.22-1 summarizes the emission of radioactivity to the atmosphere from the three Knolls Atomic Power Laboratory sites in 1977.

3.22.4 Health Impact Assessment of Knolls Atomic Power Laboratory

Tables 3.22-2 summarizes the doses from emissions of radioactivity from Knolls Atomic Power Laboratory operations. The estimated doses reported in this table include the doses from both airborne and liquid radioactive emissions. The monitoring report did not describe the nearest population group nor indicate where the maximum individual dose occurred.

The lifetime risks of fatal cancer to the maximum exposed individual at each site and to the average individual in the region are summarized in table 3.22-3. The total number of fatal cancers per year of operation of all Knolls Atomic Power Laboratory sites is estimated to be 3.1E-5. Table 3.22-3 also summarizes the fatal cancers estimated to occur in the population within 80 kilometers of each site.

Table 3.22-1. Atmospheric emissions of radionuclides from Knolls Atomic Power Laboratory, 1977 (DOE77b)

Facility	Radionuclide	Emissions (Ci/y)
Knolls Site	Argon-41	4.1
	Iodine-131	1.7E-5
	Krypton-85	4.3E-1
	Krypton-85m	3.2E-3
	Krypton-87	6.6E-3
	Krypton-88	1.1E-2
	Krypton-89	6.2E-4
	Mixed Fission Products	1.2E-4
	Plutonium-239	1.3E-8
	Antimony-125	3.6E-4
	Thorium-232	1.4E-6
	Uranium-235	2.9E-6
	Xenon-131m	2.3E-4
	Xenon-133	1.2E-3
	Xenon-135	9.7E-3
	Xenon-138	2.8E-3
Kesselring Site	Argon-41	8.5
	Carbon-14	1.0E-1
	Cobalt-60	1.0E-5
	Krypton-83m	4.2E-3
	Krypton-85	1.3E-5
	Krypton-85m	1.2E-2
	Krypton-87	1.3E-2
	Krypton-88	2.5E-2
	Xenon-131m	5.7E-4
	Xenon-133	1.1E-1
	Xenon-133m	3.6E-3
	Xenon-135	1.0E-1
Windsor Site	Argon-41	6.8E-1
	Carbon-14	6.3E-3
	Cobalt-60	3.8E-6
	Krypton-83m	9.4E-4
	Krypton-85	2.9E-6
	Krypton-85m	2.8E-3
	Krypton-87	3.0E-3
	Krypton-88	5.9E-3
	Xenon-131m	1.3E-4
	Xenon-133	2.4E-2
	Xenon-133m	8.4E-4
	Xenon-135	2.5E-2

Table 3.22-2. Annual radiation doses[a] from atmospheric emissions of radionuclides from Knolls Atomic Power Laboratory, 1977 (GE78)

Site	Maximum individual Site boundary (mrem/y)	Maximum individual Nearest population (mrem/y)	Population (person-rem/y)
Knolls Site	1.0E-2	2.0E-4	3.7E-2
Kesselring Site	4.0E-2	4.0E-4	1.1E-1
Windsor Site	1.0E-3	2.0E-5	8.0E-3

[a] Total body dose from airborne and liquid emissions.

Table 3.22-3. Individual lifetime risks and number of fatal cancers due to radioactive atmospheric emissions from Knolls Atomic Power Laboratory, 1977

Site	Individual lifetime risks Maximum individual	Individual lifetime risks Average individual Region[a]	Expected fatal cancers per year of operation (Fatal cancers)
Knolls Site	1.4E-7	4.2E-10	7.4E-6
Kesselring Site	5.6E-7	1.4E-09	2.2E-5
Windsor Site	1.4E-8	3.6E-11	1.6E-6

[a] The region extends to 80 kilometers.
[b] To the population within 80 kilometers.

REFERENCES

DOE77a Department of Energy, 1977, Effluent Information System Report No. 02, Narrative Summary Data Base Master List, EIS 02, (Computer Listing).

DOE77b Department of Energy, 1977, Effluent Information System Report No. 51, Release Point Analysis Report for Calendar Year 1977, EIS 51, (Computer Listing).

GE78 General Electric Company, 1978, Knolls Atomic Power Laboratory Annual Environmental Monitoring Report Calendar Year, 1977, KAPL-M-7537, Schenectady, N.Y.

3.23 Shippingport Atomic Power Station

3.23.1 General Description

The Shippingport Atomic Power Station, operated by the Duquesne Light Company for the Department of Energy, was the first large scale central station nuclear reactor in the United States. Initial power generation was achieved in December 1957. In 1977 the station was shut down for the installation of the light water breeder reactor (LWBR) core; initial criticality was achieved in August 1977 and full power in September 1977.

The Shippingport Atomic Power Station is located on the same site as the Beaver Valley Power Station, also operated by the Duquesne Light Company. The site is a 197-hectare tract of land located along the Ohio River in the Borough of Shippingport, Beaver County, Pennsylvania. The site is approximately 40 kilometers northwest of Pittsburgh. Beaver County, Pennsylvania, is considered an integral part of the greater Pittsburgh industrial complex. There are approximately 3.8 million people living within 80 kilometers of the site.

3.23.1 Process Description

The nuclear reactor at Shippingport Atomic Power Station is a pressurized water reactor (PWR); however, it has the LWBR core which operates on the basis of the thorium fuel cycle. Typical characteristics of the plant and LWBR core are summarized in table 3.23-1. The reactor fuel is in the form of ceramic fuel pellets with uranium-233 as the fissile material and thorium-232 the fertile material. The major difference in operation of the LWBR core from previous PWR cores, other than the type of fuel, is that the reactivity behavior of the LWBR core is controlled by movable seed or fissile fuel elements rather than by traditional control rods.

The potential source of radioactive airborne emissions is the reactor coolant system which contains activated corrosion and wear products, activated impurities, and small quantities of fission products. The radioactivity can be released and become airborne from coolant leaks, sampling operations, and maintenance and overhaul operations.

Gaseous wastes stripped from the reactor coolant are circulated through a hydrogen analyzer and catalytic hydrogen burner system where the hydrogen is removed. The gases are initially stored in a vent gas surge drum, sampled, and subsequently compressed and transfered to one of four gas storage drums. After a long decay period, the decayed gases are sampled again prior to release. In addition, the exhaust from the containment is equipped with high efficiency filters to prevent release of radioactive particulates. Protective devices are utilized in the event of high airborne activity to automatically seal off the primary containment to prevent an inadvertent release of radioactivity. Reactor plant exhausts from the Decontamination Room, Sample Preparation Room, Laundry Room, Radiochemistry Laboratory, Gaseous Waste System, and Compacting Station are also equipped with high efficiency filters and are continuously monitored for radioactive particulate by the use of fixed filter monitors.

2.23.3 Emissions of Radionuclides

Table 3.23-2 summarizes the emissions from Shippingport Atomic Power station in 1977. Since the plant was not in operation the entire year, table 3.23-3 is provided to summarize the estimated emissions to the atmosphere for the year.

Table 3.23-1. Major core and plant parameters for the LWBR core at Shippingport Atomic Power Station (ERDA76)

Parameter	Value
POWER PLANT	
Gross electrical output, MW(e)	[a]62
Net station output, MW(e)	[a]50
Net station heat rate, Btu/kW-hr	13,450
Steam pressure	
Full load at generator, psia	744
No load at generator, psia	895
Number of loops	4
Reactor pressure drop, psi	69.2
Coolant piping, OD, in.	18
Coolant piping, ID, in.	15
Coolant velocity, main piping, ft/sec	35
REACTOR CORE	
Type	Pressurized light water cooled and moderated seed and blanket
Total reactor heat output, MW(t)	[a]204
Total coolant flow rate, 10^6 lb/hr	30.6
Reactor coolant inlet temperature at 236.6 MW(t), °F	520
Reactor coolant outlet temperature at 236.6 MW(t), °F	542
Average coolant temperature, nominal , °F	531
Primary system pressure, nominal, psia	2000
Nominal core height, including ThO_2 reflector, ft	10.0
Mean core diameter, ft	7.5
Fuel loading (thorium and uranium), metric tons	~42
Lifetime, EFPH	15,000[a]

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[a] See footnotes at end of table.

Table 3.23-1. Major core and plant parameters for the LWBR core at Shippingport Atomic Power Station--continued

Parameter	Value
Fuel material	
Movable Seed	$^{233}\mathrm{UO_2}$-$\mathrm{ThO_2}$; with $\mathrm{ThO_2}$ end reflectors
Stationary Blanket	$^{233}\mathrm{UO_2}$-$\mathrm{ThO_2}$; with $\mathrm{ThO_2}$ end reflectors
Reflector Blanket	$\mathrm{ThO_2}$
Fuel cladding material	
Seed, blanket, and reflector	Zircaloy-4, low Hafnium

[a] These are the minimum expected performance values for the LWBR operation at Shippingport. To assure that environmental impacts have been conservatively evaluated, the following parameters have been analyzed in the environmental statement:

Gross electrical output, MW(e)	72
Net Station output, MW(e)	60
Total reactor heat output, MW(t)	236.6
Lifetime, EFPH	18,000

Table 3.23-2. Emissions of radionuclides to the atmosphere, Shippingport Atomic Power Station, 1977 (DOE77b)

Radionuclide	Emissions (Ci/y)
Cobalt-60	2.5E-6
Manganese-54	1.6E-8
Xenon-133	1.8E-4

Table 3.23-3. Estimated annual emissions of radionuclides from LWBR operations (ERDA76)

Radionuclide	Emissions (Ci/y)
Argon-41	2.4
Krypton-83m	1.1E-2
Krypton-85m	1.8E-2
Krypton-85	2.5E-6
Krypton-87	3.3E-2
Krypton-88	5.7E-2
Iodine-130	7.0E-7
Iodine-131	6.4E-5
Iodine-132	4.6E-3
Iodine-133	1.0E-3
Iodine-134	1.1E-2
Iodine-135	2.4E-3
Xenon-131m	3.0E-8
Xenon-133m	6.1E-4
Xenon-133	6.0E-2
Xenon-135	1.9E-1
Xenon-135m	4.9E-2
Xenon-137	1.8E-1
Xenon-138	1.5E-2

3.23.4 Health Impact Assessment of Shippingport Atomic Power Station

The maximum exposure to a hypothetical individual residing at the site boundary would be less than 0.5 mrem from airborne radioactive emissions in 1977. These same emissions result in a population dose of less than 1.0 person-rem to the population living within 80 kilometers. Table 3.23-4 and table 3.23-5 summarize the doses and risks from radioactive airborne emissions from the Shippingport Atomic Power Station in 1977.

Since Shippingport Atomic Power Station was not in full operation for the entire year of 1977 and because it shares the same site as the Beaver Valley nuclear power plants, table 3.23-6 and table 3.23-7 are provided for comparison.

Table 3.23-4. Annual radiation doses from radioactive airborne emissions from Shippingport Atomic Power Station, 1977 (DLC78)

Organ	Maximum individual dose at the site boundary (mrem/y)	Population dose within 80 km (Person-mrem/y)
Total Body	0.5	0.1

Table 3.23-5. Individual lifetime risks and number of fatal cancers from radioactive emissions, Shippingport Atomic Power Station, 1977

Source	Individual lifetime risks		Expected fatal cancers per year of operation (Fatal Cancers)
	Site boundary	Average individual Region	
Shippingport Atomic Power Station	7.0E-6	3.7E-9	2.0E-4

Table 3.23-6. Estimated annual radiation doses to the maximum individual at the site boundary from radioactive airborne emissions, Shippingport Atomic Power Station (ERDA76)

Organ	From Shippingport only (mrem/y)	Combined effect from Shippingport and Beaver Valley Units 1 & 2 (mrem/y)
Bone	1.8E-2	4.7E-1
G.I. Tract	1.8E-2	4.7E-1
Thyroid	3.7E-2	7.7
Skin	2.7E-2	8.3E-1
Total Body	1.8E-2	4.7E-1

Table 3.23-7. Estimated population doses from Shippingport and the combined effects of Shippingport and Beaver Valley Units 1 and 2 (ERDA76)

Facility	Average individual (mrem/y)	Population[a] (person-rem/y)
Shippingport	8.0E-4	3.2
Shippingport and Beaver Valley 1 & 2	1.0E-3	4.0

[a] To the population within 80 km.

REFERENCES

DOE77a Department of Energy, 1977, Effluent Information System Report No. 02, Narrative Summary Data Base Master List, EIS 02, (Computer Listing).

DOE77b Department of Energy, 1977, Effluent Information System Report NO. 51, Release Point Analysis report for Calendar Year 1977, EIS 51, (Computer Listing)

DLC78 Duquesne LIght Company, 1978, 1977 Environmental Report, Radiological - Volume #2, Duquesne LIght Company, Beaver Valley Power Station and Shippingport Atomic Power Station.

ERDA76 Energy Research and Development Agency, 1976, Final Environmental Statement, Light Water Breeder Reactor Program, ERDA-1541, Washington, D.C.

3.24 Reactive Metals, Inc., Company (RMI)

3.24.1 General Description

The Reactive Metals, Inc., Company (RMI) operates a uranium extrusion plant for the formation of rod or tubing from uranium ingots for use in reactor fuel elements.

The RMI Company plant is located in Ashtabula, Ohio, in the northeastern corner of the State.

3.24.2 Process Description

There are four stacks that emit airborne radioactivity from the extrusion processes. The operations that are associated with these release points are: Stack 1 exhausts from the extrusion press tunnel where ingots are converted to rod or tubing; Stack 2 exhausts from the extrusion exit roundout table for the press; Stack 4 emits air from the abrasive saw where the extrusions are sectioned; and Stack 5 exhausts where the pyrophoric scraps are converted to oxide. Stack 5 is the only stack equipped with a waste treatment system. Exhausted gases are passed through a Roto-Clone Type N Air Scrubber before release to the atmosphere. All stacks are sampled on a regular basis.

3.24.3 Emissions of Radionuclides

Table 3.24.1 summarizes atmospheric emissions of radionuclides from the RMI Company in 1977. Natural uranium is the only radionuclide reported.

3.24.4 Health Impact Assessment of the RMI Company

There were no reported doses for this facility.

Table 3.24-1. Atmospheric emissions of radionuclides, RMI Company, 1977

Stack number	Radionuclide	Emissions (Ci/y)
1	Uranium-238	6.7E-4
2	Uranium-238	1.1E-4
4	Uranium-238	2.0E-2
5	Uranium-238	5.9E-4
Total		2.1E-2

REFERENCES

DOE77a Department of Energy, 1977, Effluent Information System Report No. 02, Narrative Summary Data Base Master List, EIS 02, (Computer Listing).

DOE77b Department of Energy, 1977, Effluent Information System Report No. 51, Release Point Analysis report for Calendar Year 1977, EIS 51, (Computer Listing).

3.25 Lawrence Berkeley Laboratory

3.25.1 General Description

Lawrence Berkeley Laboratory (LBL) is a large multidisciplinary research institute. The laboratory carries out a wide range of programs of research in the fields of physical and biological sciences. The facilities include a number of large accelerators, and various physics, chemistry, biology and medical research laboratories.

The Lawrence Berkeley Laboratory (LBL) is part of the University of California and is located on the western slope of the hills parallel to the eastern side of San Francisco Bay. Populated residential areas of the cities of Berkeley and Oakland enclose the site to the north and south. The Berkeley Campus of the University of California is on the west of LBL, and an uninhabited regional park is to the east of the site. There are about 4.5 million people residing within 80 kilometers of LBL.

3.25.2 Process Description

There are four large high energy particle accelerators at LBL. The Bevatron, the SuperHilac and the 88-Inch Cyclotron are in almost continuous operation while the 184-inch Synchrocyclotron is used for short periods for biomedical studies.

The Bevatron is a large proton synchrotron, used for physics research requiring energies up to 6.2 GeV. The SuperHilac is a linear accelerator capable of accelerating natural elements, including uranium, up to energies of about 8 MeV per nucleon or a maximum of nearly 2 GeV per particle. The SuperHilac is used to study transuranic elements and as an injector to the Bevatron.

When used as an injector to the Bevatron, a hybrid accelerator is formed called the Bevalac. With this instrument heavy ions may be accelerated to several GeV per nucleon and applied to research in high energy physics, nuclear chemistry, radiobiology and radiotherapy. The 88-inch sector-focused cyclotron accelerates light and medium mass nuclei to energies intermediate between the SuperHilac and the Bevalac. It is used for studies of nuclear structure and radioisotope production.

The use of radionuclides in various research laboratories is the principal potential source of leakage of radionuclides into the environment. There are over 100 such exhaust points, located on a number of different buildings throughout the site. Most of these consist of chemical laboratory room exhausts. Each laboratory room has its own locally controlled exhaust. Handling of significant quantities of radioactive materials is conducted in glove boxes which are equipped with HEPA filters.

3.25.3 Emissions of Radioanuclides

The total quantities of radionuclides emitted into the atmosphere are summarized in table 3.25-1. These emissions were from chemical laboratory research operations and not from accelerator operations.

Table 3.25-1. Atmospheric emissions of radionuclides from Lawrence Berkeley Laboratory, 1977 (DOE77b, LBL78)

Radionuclide	Emissions (Ci/y)
Tritium	7.8E+1
Carbon-14	2.5E-1
Gallium-67	1.3E-3
Iodine-125	4.6E-4
Unidentified Alpha	<1.0E-6
Unidentified Beta & Gamma	4.1E-5

3.25.4 Health Impact Assessment of Lawrence Berkeley Laboratory

Maximum individual doses from radioactive airborne emissions were not reported.

Airborne emissions resulted in a dose of 2.88 person-rem to the population within 80 kilometers. Table 3.25-2 summarizes the individual lifetime risks and number of fatal cancers to the population from these doses.

Table 3.25-2. Population dose, individual lifetime risks, and number of fatal cancers due to radioactive atmospheric emissions from Lawrence Berkeley Laboratory, 1977

Radio-nuclides	Population dose (person-rem/y)	Individual lifetime risks Average individual Region[a]	Expected fatal cancers per year of operation (Fatal cancers)
Tritium	2.4	7.5E-09	4.8E-4
Carbon-14	.02	6.2E-11	4.0E-6
Iodine-125	.09	2.8E-10	1.8E-5
Gallium-67	.04	1.2E-10	8.0E-6
Unknown Beta and Gamma	.03	9.3E-11	6.0E-6
Unknown Alpha	.3	9.3E-10	6.0E-5
Total	2.88	9.0E-09	5.8E-4

[a] The region extends to 80 kilometers.

[b] To the population within 80 kilometers.

REFERENCES

DOE77a Department of Energy, 1977, Effluent Information System Report No. 02, Narrative Summary Data Base Master List, EIS 02, (Computer Listing).

DOE77b Department of Energy, 1977, Effluent Information System Report No. 51, Release Point Analysis Report for Calendar Year 1977, EIS 51, (Computer Listing).

ERDA Energy Research and Development Administration, 1976, Environmental Monitoring at Major U.S. Energy Research & Development Administration Contractor sites, Calendar Year 1975, ERDA-76-104, Washington, D.C.

LBL78 Lawrence Berkeley Laboratory, 1978, Annual Environmental Monitoring Report of the Lawrence Berkeley Laboratory 1977, LBL 7570, UC-41, Berkeley, California.

3.26 Fermi National Accelerator Laboratory

3.26.1 General Description

Fermi National Accelerator Laboratory (Fermilab) is a proton synchrotron that can operate at energies up to 500 GeV. Operations at 400 GeV are now routine. The primary purpose of the installation is fundamental research in high energy physics.

Fermilab is located in the greater Chicago area on a 2750-hectare tract of land east of Batavia, Illinois. The area around the site is changing from farming to residential use with many municipalities in the vicinity. About 8 million people live within 80 kilometers of the site.

3.26.2 Process Description

The proton beam is extracted from the 2 km diameter main accelerator and is taken to three different experimental areas. Radioactivity is produced as a result of the interaction of the accelerated protons with matter. Airborne radioactivity is produced by radioactivation of air when the proton beam or the spray of secondary particles resulting from its interactions with matter passes through air. Most proton beam lines travel inside evacuated pipes; thus, radioactivation of the air is usually caused by secondary particles. No effluent treatment system is used, but monitoring of such activation is carried out to control occupational exposure.

3.26.3 Emissions of Radionuclides

Radioactive gas, primarily carbon-11, is produced by the interaction of secondary particles with air. Small amounts of radioactivity were released from stacks in the Neutrino Area during 1977. Tritiated helium was also released from the Meson Area Target Box. Table 3.26-1 summarizes the emissions from Fermilab in 1977.

3.26.4 Health Impact Assessment of Fermilab

The maximum dose at the site boundary from radioactive airborne emissions was 0.3 mrem. Airborne emissions resulted in an off-site exposure of 0.5 person-rem to the population within 80 kilometers. Tables 3.26-2 and 3.26-3 summarize the doses and risks resulting from Fermilab operations.

Table 3.26-1. Atmospheric emissions of radionuclides from Fermi National Accelerator Laboratory, 1977

Source	Radionuclide	Emissions (Ci/y)
Neutrino Area	Carbon-11	1.0E+4
Meson Area Target Box	Tritium	5.5E-1

Table 3.26-2 Annual radiation doses from radioactive airborne emissions from Fermi National Accelerator Laboratory, 1977

Organ	Maximum individual Site boundary (mrem/y)	Population within 80 km (person-rem/y)
Total body	0.3	0.5

Table 3.26-3. Individual lifetime risks and number of fatal cancers due to radioactive emissions from Fermi National Accelerator Laboratory

Source	Individual lifetime risks		Expected fatal cancers per year of operation (Fatal Cancers)
	Maximum individual Site boundary	Average individual Region[a]	
Fermilab	4.2E-6	8.8E-10	1.0E-4

[a] To the population within 80 km.

REFERENCES

Ba78 Baker, Samuel I., 1978, Environmental Monitoring Report for Calendar Year 1977, Fermilab-78/27, 1104.100, Fermi National Accelerator Laboratorey, Batavia, Illinois, .

3.27 Stanford Linear Accelerator Center

3.27.1 General Description

Stanford Linear Accelerator Center (SLAC) is a large research laboratory devoted to theoretical and experimental research in high energy physics and to the development of new techniques in high accelerator particle detectors.

SLAC is located about 3 kilometers west of the Stanford campus in San Mateo County, California. The total length of the accelerator and the experimental area is approximately 4.8 kilometers, oriented almost east-west. The accelerator center occupies about 170 hectares of Stanford University land in the foothills of the Santa Cruz Mountains on the San Francisco peninsula. The site is halfway between San Francisco and San Jose. There are about 4.2 million people living in the six county area of the San Francisco Bay Area.

3.27.2 Process Description

The linear accelerator at Stanford is 2 miles long. It is capable of accelerating beams of electrons with energies up to 22 billion electron volts (GeV), and positrons up to 15 GeV. The east end of the 2-mile accelerator contains the research area. Included in this area are the beam switchyard, end stations A and B, counting house, data, assembly, and cyrogenics buildings and several utility buildings.

The accelerator and beam switchyard are vented at a location slightly above roof level after the electron beam is shut off. The area is vented for 10 minuits before entry.

3.27.3 Emissions of Radionuclides

Release of airborne radioactivity is infrequent and only for brief periods of time, usually 30-60 minutes. During 1977 only 1.7 curies of short-lived gaseous radioactivity were released to the atmosphere from SLAC. The isotopes emitted have half-lives ranging from 2 minutes to 1.8 hours. The isotopes are oxygen-15, nitrogen-13, carbon-11, and argon-41.

3.27.4 Health Impact Assessment of Stanford Linear Accelerator Facility

The maximum individual dose from airborne radioactive emissions from SLAC was less than 0.03 mrem per year. This may be compared to a maximum annual dose at the site boundary of 8.2 mrems from penetrating neutron radiation. No population dose was reported from airborne emissions from Stanford Linear Accelerator operations. Tables 3.27-1 and 3.27-2 summarize the dose and lifetime risks associated with airborne emissions from SLAC.

Table 3.27-1. Annual radiation doses from radioactive airborne emissions from Stanford Linear Accelerator Center, 1977

Source	Maximum individual Site boundary (mrem/y)	Population (person-rem/y)
SLAC	<0.03	NR

NR Not reported.

Table 3.27-2. Individual lifetime risks and number of fatal cancers due to radioactive emissions from Stanford Linear Accelerator Center

Source	Individual lifetime risks Maximum individual Site boundary	Expected fatal cancers per year of operation (Fatal Cancers)
SLAC	<4.2E-7	NR

REFERENCES

ERDA76 Energy Research and Develoment Administration, 1976, Final Environmental Statement, Positron-Electron Storage Ring Project, Stanford Linear Accelerator Center, Stanford, California, ERDA-1546, Washington, D. C.

SLAC78 Stanford Linear Accelerator Center, 1978, Annual Environmental Monitoring Report, January-December 1977, Stanford, California.

CHAPTER 4

SOURCES OF EMISSIONS OF NATURALLY OCCURRING RADIONUCLIDES

This chapter deals with sources which emit naturally occurring radioactive materials into the atmosphere. The source categories include: uranium mining, uranium milling, phosphate mining and processing, coal-fired steam electric generation, metal and non-metal mining (other than uranium and phosphate), and underground water sources. These are the source categories, which at the present time, have been identified as having the greatest potential for release of naturally occurring radionuclides to the air. Future reports will include the assessments of other source categories now being evaluated. The uranium mining and milling source categories could have been included in Chapter 2 under the uranium fuel cycle; however, they are presented in this chapter because the types of activities and radionuclides involved with these categories are similar to those treated in the other sections of this chapter.

Naturally occurring radionuclides fall into two general categories: primordial and cosmogenic. The more important of these, in terms of air emissions, are the primordial radionuclides and their daughter products. These would include the uranium-238, uranium-235, and thorium-232 decay series. The soils and rocks which make up the earth's crust contain these radionuclides and their daughter products in widely varying amounts. Average values for uranium-238 and thorium-232 in soils have been reported to be about 1.8 ppm (0.6 pCi/g) and 9 ppm (1 pCi/g), respectively (NCRP75).

Because of the presence of these radionuclides in the earth's crust, almost all of man's activities which involve the removal and processing of materials from the earth's surface, or the removal of gases, vapors, or liquids from below the earth's surface can result in the release of some of these radioactive materials to the atmosphere. These releases can become potentially important when (1) the activity involves the handling of materials containing concentrations of these radionuclides (specific activities) significantly above the average concentrations in soil, (2) these radionuclides are concentrated during processing to a level significantly above the average concentrations in soil, or (3) the radioactive material is redistributed from its place in nature into a pathway where man can be exposed. Each of the source categories covered in this chapter involves one or more of the above considerations.

REFERENCES

NCRP75 National Council on Radiation Protection and Measurements, 1975, Natural Background Radiation in the United States, NCRP Report No. 45, Washington, D. C.

4.1 Uranium Mining

4.1.1 General Description

Uranium mining operations involve the removal from underground of large quantities of ore containing uranium and its daughter products in concentrations up to 1000 times the concentrations of these radionuclides in the natural terrestrial environment. The concentration of uranium in currently mined ores ranges from 0.1 to 0.2 percent U_3O_8 or 280 to 560 microcuries of uranium-238 per metric ton of ore. Since the uranium-238 in these ores is usually present in secular equilibrium with its daughter products, these ores also contain an equal amount of each of the members of the uranium decay series.

After mining, the ores are shipped to a uranium mill (see section 4.2) for separation of uranium for subsequent use in light water nuclear power reactors. Emissions from uranium mines consist of airborne radioactive dusts and radon-222 gas.

Uranium mining is generally carried out either by open pit or underground mining methods. When ore deposits are near the surface, open pit methods are used. For deep ore deposits underground methods are preferred. In 1977 there were 251 underground and 36 open pit uranium mines in operation in the United States (table 4.1-1). These mines accounted for about 96 percent of the uranium produced with each mining method accounting for approximately the same amount of uranium (DOE78). In recent years in-situ solution mining has become more widely used and the amount of uranium mined by this method is expected to increase in future years. However, during 1977 this method accounted for only a few percent of the uranium mined in the United States.

All of the present uranium mining takes place in Western States. In general these mines are located in relatively remote low population areas. Table 4.1-2 shows the production of uranium in ore by State. These data show that 77 percent of the uranium production takes place in the States of New Mexico and Wyoming.

Projections of future requirements of uranium by the nuclear power industry indicate that an annual production of about 75,000 metric tons of U_3O_8 in ore will be needed in the year 2000 (NRC79). This is about a five-fold increase in uranium in ore production over 1977 and indicates a substantial increase in uranium mining in the next two decades.

Table 4.1-1. Distribution of 1977 U_3O_8 production in ore by mining method (DOE78)[a]

Source	Number	Tons U_3O_8	% of total
Underground mines	251	8,300	50
Open pit mines	36	7,600	46
Others: heap leach, mine water, solution mining, low-grade stockpiles	27	800	4
Total	314	16,700	100

[a] Short tons

Table 4.1-2. Distribution of 1977 U_3O_8 production in ore by State (DOE78)[a]

State	Tons of ore	Tons U_3O_8	% of total U_3O_8
New Mexico	4,209,000	7,600	46
Wyoming	3,834,000	5,200	31
Others: Arizona, Colorado, Texas, Utah, & Washington	2,967,000	3,900	23
Total	11,012,000	16,700	100

[a] Short tons

4.1.2 Process Description

Underground Mining

Underground uranium mining is usually carried out using a modified room and pillar method. In this method, a large diameter main entry shaft is drilled to a level below the ore body. A haulage way is then established underneath the ore body. Vertical raises are then driven up from the haulage way to the ore body. Development drifts are driven along the base of the ore body connecting with the vertical raises. Mined ore is hauled along the development drift to the vertical raises and gravity fed to the haulage way for transport to the main shaft for hoisting to the surface.

Figure 4.1-1 is an example of an underground mining operation. Ventilation shafts are installed at appropriate distances along the ore body. Typical ventilation flow rates are on the order of 6,000 m^3/min. The principal radioactive effluent in the mine ventilation air is radon-222 which is released during mining operations.

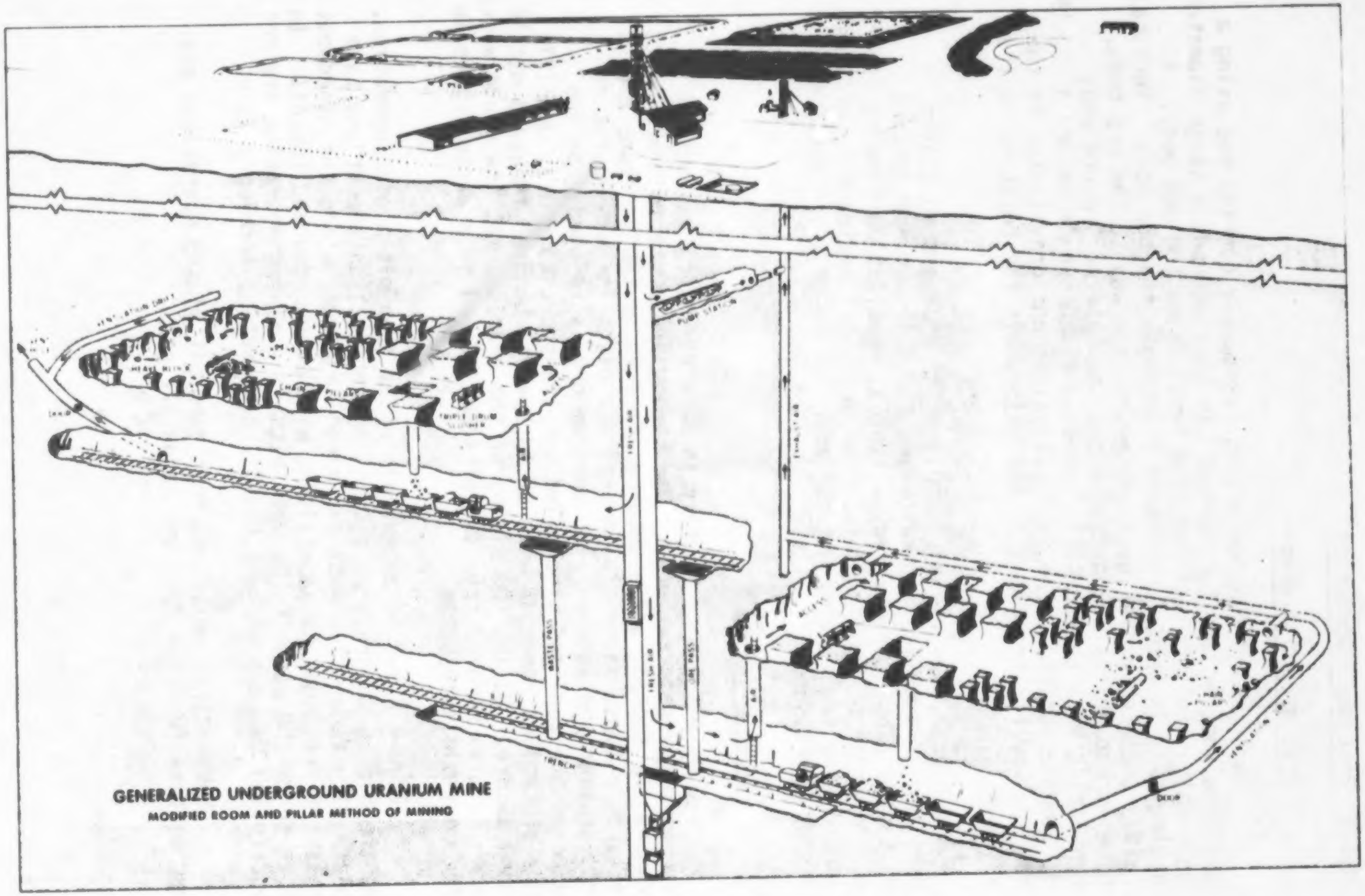
Surface Mining

Open pit mining usually is carried out by excavating a series of pits in sequence. The mining procedure followed is to remove the topsoil and overburden from above the ore zone and to stockpile these materials in separate piles for use in future reclamation operations. The uranium ore is removed from the exposed ore zone and stockpiled for transport to a uranium mill. Ore stockpiles range in size up to several hundred thousand metric tons of ore. During the mining of the uranium ore, low grade waste rock is also removed from the pits and stored in a waste stockpile for possible future use.

Figure 4.1-2 is an example of an open pit mining operation. As the mining progresses, mining and reclamation operations take place simultaneously--pits are mined in sequence and the mined-out pits are reclaimed by backfilling with overburden and topsoil. In some cases the last of the open pits in a mining operation are not backfilled but are allowed to fill with water, forming a lake.

Radioactive emissions from open pit mining operations are radioactive fugitive dust and radon-222 gas.

Figure 4.1-1. Generalized underground mine showing modified room and pillar method of mining


GENERALIZED UNDERGROUND URANIUM MINE
MODIFIED ROOM AND PILLAR METHOD OF MINING
VENTILATION DRIFT
HEAVY MINING
PILLARS
TRIPLE DRUM SLUSHER
DOOR
PUMP STATION
ACCESS
WASTE PASS
ORE PASS
FRESH AIR
TRENCH

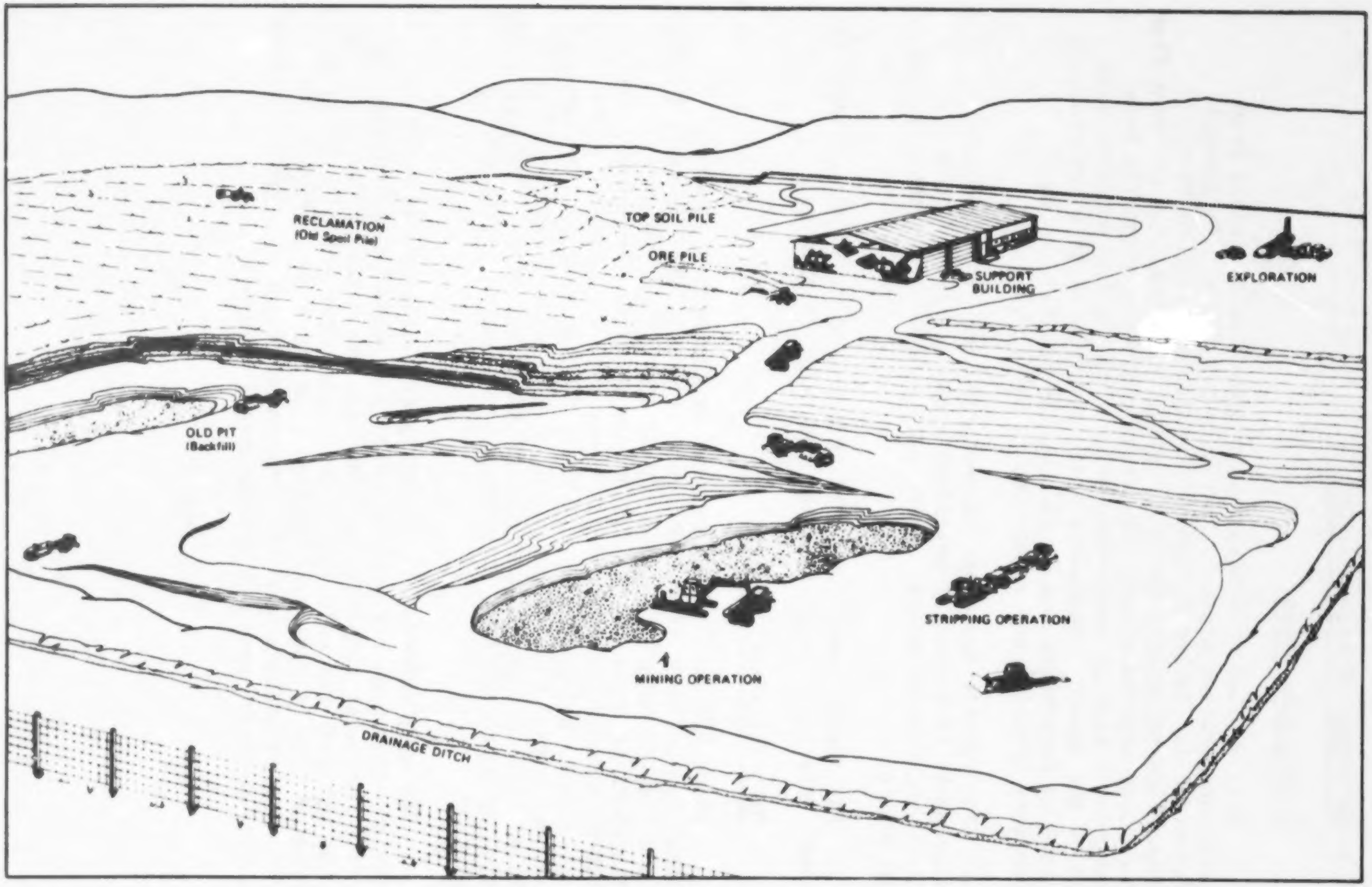

RECLAMATION
(Old Spoil Pile)
TOP SOIL PILE
ORE PILE
SUPPORT BUILDING
EXPLORATION
OLD PIT
(Backfill)
STRIPPING OPERATION
MINING OPERATION
DRAINAGE DITCH

Figure 4.1-2. Artist's conception of open pit mining operation and support facilities

4.1.3 Emissions of Radionuclides

Underground Mines

Only limited information on radioactive emissions from underground uranium mines is currently available. In general information is limited to radon-222 emissions because this radionuclide represents the major airborne radioactive effluent from underground uranium mines. The data originates primarily from measurements made at seven Kerr-McGee mines in the Ambrosia Lake area of New Mexico. Radon-222 emissions from these mines measured in 1976 by Kerr-McGee and in 1978 by the Pacific Northwest Laboratory (PNL79a) are summarized in table 4.1-3.

The Pacific Northwest Laboratories are also conducting measurements of radon-222 emissions from a series of other underground uranium mines. These data should be available in the near future.

Particulate emissions from underground mines are believed to be much less significant than the radon-222 emissions.

Open Pit Mines

Radon-222 is also believed to be the major airborne radioactive effluent from open pit uranium mining. However, because of the diffuse nature of these mining operations, no direct measurement of radon-222 emissions can be made. Therefore, emission rates of radon-222 from open pit mines must be estimated using calculational methods. These methods involve calculating radon-222 emissions either (1) from measured or estimated fluxs of radon-222 exhaled from the various surfaces of the mine, or (2) from measured radon-222 concentrations in surface air downwind of the mining operation.

The only information currently available on radon-222 emissions from open pit uranium mines are estimates calculated by Pacific Northwest Laboratories (PNL79b). An annual release of 2000 curies per year was calculated for a model open pit mine based on an estimated flux of radon-222 exhaled from the mine surface. The model mine parameters used in these calculations were obtained from a survey of eight open pit uranium mines in Wyoming.

Table 4.1-3. Radon-222 emissions from Kerr-McGee underground uranium mines at Ambrosia Lake, New Mexico

		[a]1976			
Mine section	Number of vents	Ventilation rate (m^3/min)	Ore process rate (t/d)	Radon-222 released	
				Ci/d	Ci/t
17	10	5,900	327	13.1	0.040
19	4	4,200	428	4.3	0.010
24	6	3,900	208	12.2	0.059
30	11	10,400	548	12.1	0.022
30W	5	8,100	863	4.3	0.005
35	4	9,900	867	3.7	0.004
36	4	5,300	308	5.1	0.017

		[b]1978			
Mine section	Number of vents	Ventilation rate (m^3/min)	Ore process rate (t/d)	Radon-222 released	
				Ci/d	Ci/t
17	10	5,600	328	22	0.067
19	4	4,400	420	4	0.010
24	6	4,800	208	40	0.19
30	11	11,400	548	23	0.043
30W	5	8,100	865	12	0.014
35	4	11,000	870	12	0.014
36	4	6,400	202	16	0.080

[a] Letter from W.J. Shelley of Kerr-McGee to Ralph M. Wilde of the Nuclear Regulatory Commission, December 9, 1976.

[b] Source: PNL79a.

4.1.4 Model Facilities

In order to estimate radioactive emissions and health impacts from uranium mining operations, model underground and open pit mines were developed by assigning appropriate values for the various parameters that are important in estimating emissions . These parameters which are taken directly from work reported by Pacific Northwest Laboratories (PNL79a and PNL79b) are listed in tables 4.1-4 and 4.1-5.

Mine Emissions

The atmospheric emissions of radioactive materials from the model mines are listed in tables 4.1-6 and 4.1-7. These data were obtained from PNL reports (PNL79a and PNL79b). The data presented are limited to radon-222 emissions, the only data presently available and the radionuclide representing the major airborne effluent from uranium mines.

Table 4.1-4. Model underground uranium mine

Parameter	Value[a]
Ore mining rate	500 t per day
Mine lifetime	20 years
Ore grade	0.2% U_3O_8
Number of vents	5
Surface Ore Storage[b]	3500 t
Waste storage piles[b]	
Area	11 hectares
Height	1.2 meters
Grade	0.025% U_3O_8

[a] Source: PNL79a.
[b] Average values during mine lifetime.

Table 4.1-5. Model open-pit uranium mine

Parameter	Value[a]
Ore mining rate	1600 t per day
Mine lifetime	17 years
Average ore grade	0.11% U_3O_8
Average subore grade	0.025% U_3O_8
Average overburden grade[b]	0.002% U_3O_8
Number of pits[c]	7
Pit depth	77 meters
Overburden depth	65 meters
Ore zone depth	12 meters
Overburden to ore ration	77:1
Ore to subore ratio	1:1
Total volume of pit	$5.5x10^7$-m^3
Total volume of ore	$7.1x10^5$-m^3
Area of ore storage pile	1.1 ha
Ore stockpile residence time	41 days
Area of subore pile [d]	10 hectares

[a] Source: PNL79b.

[b] Overburden initially contains 4 ppm U_3O_8 but reaches 20 ppm through relocation and mixing.

[c] The model mine consists of a series of 7 pits in various stages of excavation, mining and reclamation. A detailed description of the model mine, and the various sources contributing to radon-222 emissions is presented in PNL79b.

[d] Area of subore pile at midpoint of mine lifetime.

Table 4.1-6. Atmospheric emissions of radionuclides from the model underground uranium mine

Source of Emission	[a]Radon-222 (Ci/y)
Mine vents[b]	6500
Ore storage	12
Waste storage piles	217
Total	6729

[a] Source: PNL79a.
[b] 1300 Ci/y per vent.

Table 4.1-7. Atmospheric emissions of radionuclides from the model open pit uranium mine

Source of Emission	[a]Radon-222 (Ci/y)
Active open pit	894
New pit being excavated	148
Ore stockpile	103
Subore waste pile	163
Overburden waste pile	148
Refilled pits	391
Increased land area	15
Truck loading and dumping	99
Total	[b]1961

[a] Source: PNL79b.
[b] Average annual radon-222 emission over the lifetime of mine.

4.1.4 Health Impact Assessment of Model Uranium Mines

The estimated working level exposures that would result from radon-222 emissions from the model uranium mines are listed in table 4.1-8. These are estimates for a low population density, generic uranium mining and milling site in the Western United States (Site E, Appendix A). The model underground mine consists of 5 vents each releasing an equal amount of radon-222. The working level exposure to the highest group of individuals was calculated for a location 500 meters from one of the vents in the predominent wind direction. The model underground mine was treated as an area source of 1000 hectares and the working level exposure to the highest exposed group of individuals was calculated at a location 500 meters from the edge of the mining area in the predominant wind direction.

In addition to the working level exposures from the inhalation of short-lived radon-222 daughter products listed in table 4.1-8, radiation doses from radon-222 emissions also occur to body organs from the inhalation and ingestion of lead-210 formed from the decay of the released radon-222. Data on the doses to the population of the United States from lead-210 from radon-222 released from uranium milling operations is presented in detail in ORNL79. Our preliminary evaluation of these data indicated that the health impact resulting from the organ doses from lead-210 is smaller than the health impact from the inhalation of the short-lived radon-222 daughters. The health risk data presented in this report does not include these small additional incremental risks from lead-210.

Estimates of the individual lifetime risks and number of fatal cancers resulting from these working level exposures are given in table 4.1-9. The risks from the model underground mine are greater than the open pit mine because of the larger quantity of radon-222 released.

For the model underground mine the lifetime risk to the highest exposed group of individuals is estimated to be about 1×10^{-2}. The lifetime fatal cancer risk to the average individual in the regional and the United States populations is estimated to be about 5×10^{-5} and 3×10^{-8} respectively. The number of fatal cancers per year of mine operation is estimated to be 0.03 in the regional population and 0.08 in the population of the United States.

Table 4.1-8. Working level exposures from radon-222 emissions from model uranium mines

Source	Maximum individual (WL)	Regional population (person-WL)	United States population[a] (person-WL)
Underground mine	[b]6.0E-3	1.3	3.8
Open pit mine	[c]8.4E-4	3.8E-1	1.1

[a] Calculated from data in ORNL79--Table 2.1 where a one kCi radon-222 release is estimated to result in an exposure of 8.0E+4 person-pCi/m^3 to the population of the United States. This is equivalent to an exposure to 0.56 person-working levels based on an assumption of a 70 percent equilibrium of the radon-222 daughter products (100 pCi/L radon-222 = 0.7 WL) which is considered to be representative of indoor exposure conditions (Ge78).

[b] Exposure to an individual living 500 meters from a mine vent in predominant wind direction

[c] Exposure to an individual living 500 meters from edge of mining area which has been treated as an area source of 1000 ha.

Table 4.1-9 Individual lifetime risks and number of fatal cancers in the population from radon-222 emissions from model uranium mines.

Source	Individual lifetime risks		
	Maximum	Average individual	
	Individual	Region	United States
Underground mine	9.0E-3	5.3E-5	2.6E-8
Open pit mine	1.3E-3	1.6E-5	7.4E-9

Source	Expected fatal cancers per year of operation		
	Region (Fatal cancers)	United States (Fatal cancers)	Total (Fatal cancers)
Underground mine	2.7E-2	8.0E-2	1.1E-1
Open pit mine	8.0E-3	2.3E-2	3.1E-2

REFERENCES

DOE78 Department of Energy, 1978, Statistical Data of the Uranium Industry, GJO-100(78), Washington, D.C.

Ge78 George A.C., and Breslin, A.J., 1978, The Distribution of Ambient Radon and Radon Daughters in Residential Buildings in the New Jersey--New York Area, Presented at Symposium on the Natural Radiation Environment III, Houston, Texas.

NRC76 Nuclear Regulatory Commission, 1976, Final Generic Environmental Statement on the Use of Recycle Plutonium in Mixed Oxide Fuel in Light Water Cooled Reactors (GESMO), NUREG-002, Vol 3, Washington, D.C.

NRC79 Nuclear Regulatory Commission, 1979, Draft Generic Environmental Impact Statement on Uranium Milling, NUREG-0511 Washington, D.C

PNL79a Jackson, P., et al., 1979, Radon-222 Emissions in Ventilation Air Exhausted from Underground Uranium Mines, PNL-2888, NUREG-CR-0627, Richland, Washington.

PNL79b Neilson K., 1979, Prediction of Net Radon Emission from a Model Open Pit Uranium Mine, PNL-2889, NUREG-CR-0628, Richland, Washington.

ORNL79 Travis C.C., et al., 1979, A Radiological Assessment of Radon-222 Released from Uranium Mills and other Natural and Technologically Enhanced Sources, (NUREG/CR-0573), ORNL/NUREG-55 Oak Ridge, Tennessee.

4.2 Uranium Mills

4.2.1 General Description

Uranium milling operations involve the handling and processing of large quantities of ore containing uranium and its daughter products in concentrations up to 1000 times the concentrations of these radionuclides in the natural terrestrial environment.

The concentration of uranium in the ores currently being processed ranges from about 0.1 to 0.2 percent U_3O_8 or 280 to 560 microcuries of uranium-238 per metric ton of ore. Since the uranium in these ores is usually present in secular equilibrium with its daughter products, these ores also contain an equal amount of each of the members of the uranium-238 decay series.

The function of a uranium mill is to extract uranium in concentrated form from naturally occurring ore deposits. The product is a semirefined uranium compound called yellowcake which is shipped to a conversion plant for a further purification step in preparing the uranium for use in light-water nuclear power reactors. Liquid and solid wastes are impounded near the mill in a tailings pond or pile. Emissions consist of airborne radioactive dusts of ore, tailings, and yellowcake and radon-222 gas.

As of January 1979 there were 20 active (conventional) uranium mills in the United States (GJO79) located in low population density areas in Western States. Table 4.2-1 lists these mills, their locations, and normal processing capacities.

In 1977 uranium mills processed about 10 million tons of ore with an average ore grade of 0.15 percent U_3O_8 containing about 4000 curies of each of the members of the uranium-238 decay series. The operation of these uranium mills has resulted in the accumulation of large quantities of waste tailings.

There are currently over 100 million tons of tailings stored at these active uranium mill sites containing in excess of 50,000 curies each of thorium-230, and radium-226 and its decay products. These tailings represent the major source of radon-222 emissions at a uranium mill site.

Projections of future uranium requirements (NRC79a) indicate that by the year 2000 about 75,000 metric tons of U_3O_8 will be needed to support the nuclear power industry. To produce this quantity of uranium would require an additional 43 uranium mills with 1800 t/day capacities. Based on this projection, a large increase in the uranium milling industry can be expected over the next two decades.

Table 4.2-1. Location and capacity of uranium mills in the United States (GJ079)

State & Company	Location	Nominal capacity ore/day (metric tons)
New Mexico		
Anaconda Company	Grants	5,450
Kerr-McGee Nuclear Corporation	Grants	6,350
Sohio Natural Resources Co.	Cebolleta	1,450
United Nuclear Corporation	Church Rock	2,720
United Nuclear-Homestake Partners	Grants	3,090
Total		19,060
Wyoming		
Exxon, U.S.A.	Powder River Basin	2,720
Federal-American Partners	Gas Hills	860
Pathfinder Mines Corporation	Gas Hills	1,500
Pathfinder Mines Corporation	Shirley Basin	1,630
Petrotomics	Shirley Basin	1,360
Bear Creek Uranium Company	Powder River Basin	1,270
Union Carbide Corporation	Natrona County	1,090
Western Nuclear, Inc.	Jeffrey City	1,540
Total		11,970
Utah		
Atlas Corporation	Moab	1,270
Rio Algom Corporation	La Sal	680
Total		1,950
Colorado		
Cotter Corporation	Canon City	410
Union Carbide Corporation	Uravan	1,180
Total		1,590
Texas		
Conoco & Pioneer Nuclear, Inc.	Falls City	2,910
Washington		
Dawn Mining Company	Ford	410
Western Nuclear	Wellpinit	1,820
Total		2,230
Grand total		39,710

4.2.2 Process Description

Ore Storage

When ore is delivered to the mill site, it is stored on an ore storage pad with the amount stored ranging from a 10-day to a 6-month supply. Therefore, at some mill sites as much as several hundred thousand tons of ore may be stored prior to milling. Ore as delivered to the storage pads generally contains a relatively high moisture content; however, significant drying out can take place during storage. Radioactive materials are released from these storage piles through diffusion of radon-222 gas, dusting from wind erosion, and heavy equipment operation. The ore is transferred from the storage pads to the mill crushing unit using front-end loaders or bulldozers.

Milling

The process of uranium extraction involves the following steps: crushing, grinding, chemical leaching, separation of the uranium from the leach solution, precipitation, drying and packaging of yellowcake. Mill processes fall into three general types: acid-leach solvent extraction, acid-leach ion-exchange, and alkaline leach. Most mills today utilize an acid-leach solvent extraction process (figure 4.2-1).

The steps in the milling process which generate the major radioactive emissions are the front-end crushing operations and the drying and packaging of yellowcake. The other operations in the milling process are carried out in a wet state and therefore do not result in the generation of any significant airborne dust emissions.

Tailings Impoundment

Uranium mill wastes are usually stored in a tailings impoundment located on the mill site. The tailings pile is usually located in a gently sloping natural drainage area and covers an area of about 30 to 60 hectares. Tailings are discharged to the impoundment area in slurry form, at about 50 percent solids. The liquids are partially recycled to the mill or undergo natural evaporation. In the past a starter dam was constructed of native soil materials and the remainder of the dam was built from sand tailings. However, present practice is to construct the dam entirely from imported materials using an impermeable clay core. Tailings dams may be as high as 30 meters. The tailings are comprised of two fractions, sands and slimes. The sand fraction (>200 mesh) makes up about 70 per cent of the tailings and the slime fraction (<200 mesh), the remaining 30 percent. However, it has been estimated (ORNL75) that the slime fraction contains about 85 percent of the radionuclide content of the tailings.

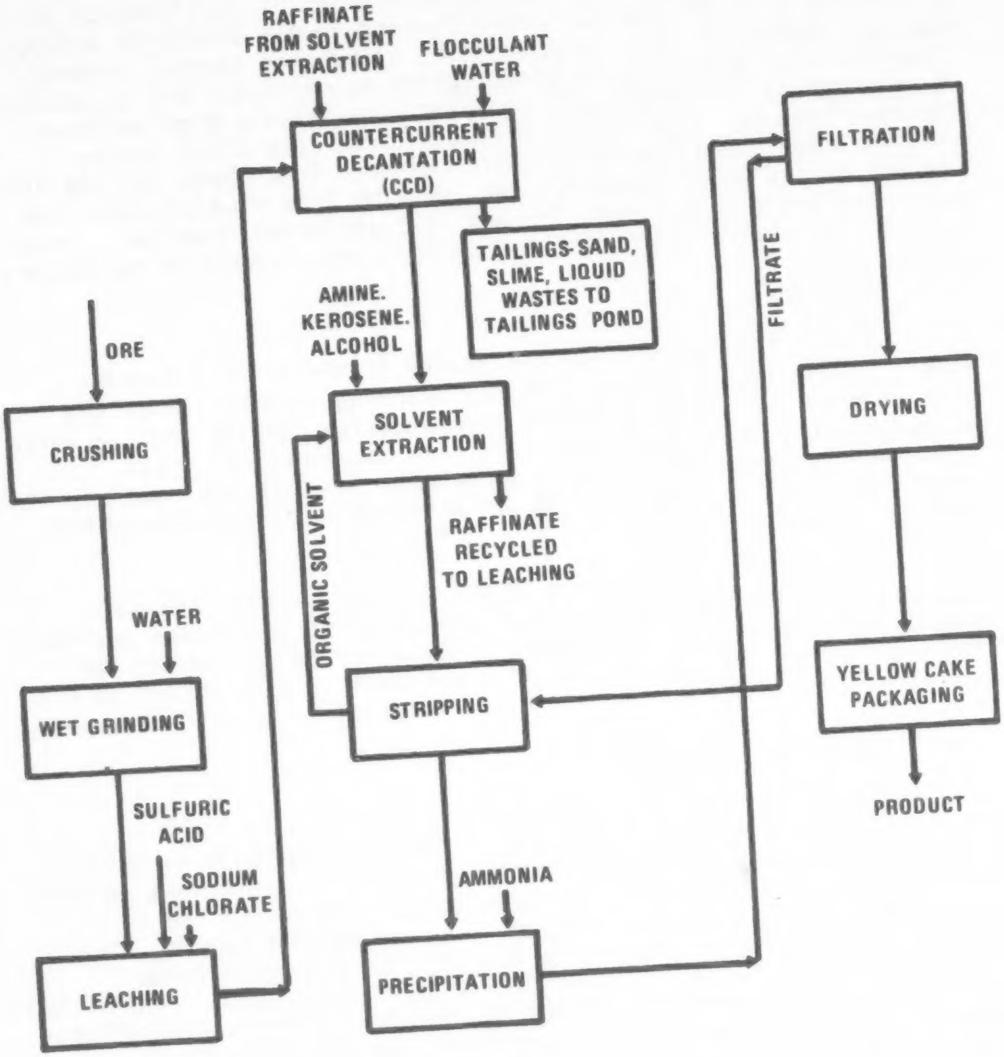

RAFFINATE FROM SOLVENT EXTRACTION
FLOCCULANT WATER
COUNTERCURRENT DECANTATION (CCD)
TAILINGS-SAND, SLIME, LIQUID WASTES TO TAILINGS POND
AMINE. KEROSENE. ALCOHOL
SOLVENT EXTRACTION
RAFFINATE RECYCLED TO LEACHING
ORGANIC SOLVENT
STRIPPING
AMMONIA
PRECIPITATION
ORE
CRUSHING
WATER
WET GRINDING
SULFURIC ACID
SODIUM CHLORATE
LEACHING
FILTRATION
FILTRATE
DRYING
YELLOW CAKE PACKAGING
PRODUCT

Figure 4.2-1. Flow diagram for the acid-leach process (NRC79a)

The tailings impoundment is made up of a pond and dry beach area with the size of each component dependent on the amount of water recycled and the rate of evaporation. In areas of high evaporation, large, dry beach areas are exposed. Radioactive airborne emissions from these dry beach areas take place as a result of wind erosion of the tailings and the diffusion of radon-222 gas. Essentially no airborne emissions originate from pond areas.

Control Technology

Methods used for controlling airborne dust emissions at uranium mills consist of wet scrubbers, impingement scrubbers, or bag filters. No attempt is made to control radon-222 emissions from these mills. Dust control of crushing and fine-bin storage areas is accomplished by passing air through a wet scrubber prior to exhausting it through a roof vent. However, a number of mills do not use exhaust ventilation in these areas--dusting is not a problem because of the high moisture content of the ore processed or the use of a semi-autogenous grinding operation. All mills utilize some method of dust control in yellowcake areas. Air cleaning systems on the yellowcake exhausts are largely wet scrubbers, usually orifice or impingement scrubbers, with Venturi scrubbers used occasionally. Bag filters are used sometimes on the packaging exhaust stack, but are not suitable for use on the dryer stack because of the temperature and moisture content of the dryer exhaust air.

4.2.3 Emissions of Radionuclides

Table 4.2-2 summarizes estimates of airborne radioactive emissions reported in the most recent environmental impact statements for uranium mills. These emission estimates vary considerably and cover a very wide range of values because of differences in the milling operations and uncertainties about actual emissions, which is due to a lack of actual measurement data on emission rates. Only a few such measurements are currently available and these represent relatively short time periods.

Emission rates for the milling operations (table 4.2-2) fall in these approximate ranges: 30-120 mCi/y for uranium-238; 30-120 mCi/y for uranium-234; 5-50 mCi/y for thorium-230, radium-226 and lead-210; and 25-170 Ci/y for radon-222. The range of emission rates from the ore storage and crushing activities is much wider than the range for the yellowcake drying and packaging activities. However, the more recent estimates for crushing and storage activities (USFS78, NRC79a, NRC79b) show a much smaller range--about 1-4 mCi/y for each of the radionuclides emitted in particulate form. The data indicate that the major source of atmospheric dust emissions from a uranium mill originate from the yellowcake drying and packaging operations.

Table 4.2-2. Estimates of radioactive emissions of uranium mills from recent environmental impact statements

Mill location and radionuclide	Emissions (Ci/y)						
	Bear Creek Project (NRC77a)	Lucky Mc Gas Hills (NRC77b)	Moab Uranium Mill (NRC77c)	Morton Ranch Uranium Mill (NRC78)	Homestake Pitch Project (USFS78)	Sweetwater Uranium Project (NRC79b)	Model Mill (NRC79a)
Ore crushing & storage							
Uranium-238	3.3E-2	5.0E-2	9.0E-3	6.3E-5	4.0E-3	1.6E-3	1.4E-3
Uranium-234	3.3E-2	5.0E-2	9.0E-3	6.3E-5	4.0E-3	1.6E-3	1.4E-3
Thorium-230	3.3E-2	5.0E-2	9.0E-3	6.3E-5	4.0E-3	1.6E-3	1.4E-3
Radium-226	3.3E-2	5.0E-2	9.0E-3	6.3E-5	4.0E-3	1.6E-3	1.4E-3
Lead-210	3.3E-2	5.0E-2	9.0E-3	6.3E-5	4.0E-3	1.6E-3	1.4E-3
Radon-222	1.2E+2	1.7E+2	5.5E+1	1.1E+2	9.2E+1	2.6E+1	1.1E+2
Yellowcake drying & packaging							
Uranium-238	3.1E-2	4.6E-2	1.0E-1	1.2E-1	2.3E-2	4.0E-2	7.2E-2
Uranium-234	3.1E-2	4.6E-2	1.0E-1	1.2E-1	2.3E-2	4.0E-2	7.2E-2
Thorium-230	1.6E-3	2.5E-3	3.0E-3	6.0E-3	-	2.0E-3	3.6E-3
Radium-226	3.1E-5	4.6E-5	1.0E-3	2.4E-4	4.6E-4	8.0E-5	1.4E-4
Lead-210	3.1E-2	4.6E-5	1.0E-3	2.4E-4	4.6E-4	8.0E-5	1.4E-4
Radon-222	-	-	-	-	-	-	-
Tailings disposal area							
Uranium-238	-	4.2E-4	2.0E-4	2.7E-3	2.3E-4	-	1.4E-2
Uranium-234	-	4.2E-4	2.0E-4	2.7E-3	2.3E-4	-	1.4E-2
Thorium-230	-	2.1E-2	4.0E-3	5.2E-2	3.3E-3	-	1.9E-1
Radium-226	-	2.1E-2	4.0E-3	5.5E-2	3.2E-3	-	2.0E-1
Lead-210	-	2.1E-2	4.0E-3	5.5E-2	3.2E-3	-	2.0E-1
Radon-222	1.6E+3	1.2E+3	5.0E+3	8.5E+3	2.3E+2	1.4E+1	7.0E+3

Emission rates from the tailings disposal area (table 4.2-2) show the following approximate ranges: 0.2-14 mCi/y of uranium-238; 0.2-14 mCi/y of uranium-234; 3-200 mCi/y of thorium-230, radium-226, lead-210; and 14-8500 Ci/y of radon-222. The amount of airborne emissions from tailings disposal areas depends upon the size of dry tailings beach areas which are subject to wind erosion and radon-222 diffusion. Tailings impoundment areas almost completely covered by water will have low radionuclide emissions (NRC79b).

As stated, actual measurement data on emissions rates are relatively scarse. However, a number of special studies are now in progress to develop information on radioactive emissions rates from uranium mills. These studies are being carried out by EPA-Las Vegas and by Argonne National Laboratory and Battelle Northwest Laboratory under contract to the Nuclear Regulatory Commission (NRC). In addition, newly developed monitoring requirements for uranium mills by the NRC (NRC77d) will result in the generation of emission data by the industry in the near future.

4.2.4 Model Facility

In order to estimate radioactive emissions and health impacts from uranium milling operations, a model uranium mill and tailings pile were developed by assigning values for the various parameters that are important in assessing impacts (table 4.2-3). The mill parameters were taken primarily from an EPA report (EPA73) and the parameters for the tailings pile are those used by Magno (Ma78).

Mill Emissions

The atmospheric emissions of radioactive material from the model uranium mill are listed in table 4.2-4. Emissions are presented separately for the milling operations and the tailings disposal area. The emission rates for the milling operations (except for radon-222) come from an EPA report (EPA76) which was based on data reported by Oak Ridge National Laboratory (ORNL75). The emission rates for the tailings disposal area were derived from information reported by Magno (Ma78) and Oak Ridge (ORNL75). The footnotes to table 4.2-4 indicate the source and basis for these emission estimates.

Table 4.2-3. Model uranium mill

Parameter	Value
Type of process	Acid-leach solvent extraction
Ore process rate	2000 metric tons per day
Operating days per year	300 days
Mill lifetime	20 years
Ore grade	0.2% U_3O_8
Uranium recovery	95%
Ore activity	560 pCi/g, Uranium-238 and daughter products in secular equilibrium
Ore storage area	1 hectare
Ore storage time	10 days
Effective stack height	15 meters
Area of tailings impoundment	60 hectares
Dry beach	15 hectares
Pond and wet beach	45 hectares
Average depth of tailings	12 meters

Table 4.2-4. Atmospheric emissions of radioactive materials from the model uranium mill

Radionuclide	Emissions (Ci/y)		
	Milling operations[a]	Tailings disposal area[b] (0-10 µm)	(10-80 µm)
Uranium-238	9.0E-2	6.0E-4	1.5E-3
Uranium-234	9.0E-2	6.0E-4	1.5E-3
Thorium-230	1.0E-2	1.2E-2	3.0E-2
Radium-226	5.0E-3	1.2E-2	3.0E-2
Lead-210	5.0E-3	1.2E-2	3.0E-2
Polonium-210	5.0E-3	1.2E-2	3.0E-2
Radon-222	[c]1.2E+2	[d]2.7E+3	

[a] Reference: EPA76--table 5.0-1.

[b] Estimated from data in ORNL75 as follows: Dust emissions for 4.5 m/s average wind speed=1.7E-2 g/ha-s for 0-10 µm particles and 4.0E-2 g/ha-s for 10-80 µm particles (table 7.4). Radionuclide concentrations of dust particles were 1610 pCi/g for ^{230}Th, ^{226}Ra, ^{210}Po, ^{210}Pb and 80 pCi/g for ^{234}U and ^{238}U (table 4.12).

[c] Based on values listed in table 4.2-1 of this report.

[d] Estimated from data in Ma78 as follows: Radon-222 release from dry tailings area = 180 Ci/ha (based on 560 pCi/g ^{226}Ra).

4.2.5 Health Impact Assessment of Model Uranium Mill

Tables 4.2-5 and 4.2-6 estimate annual radiation doses and working level exposures resulting from radioactive emissions from the model uranium mill. The estimates are for a low population density, generic uranium mining and milling site in the Western United States (Site E, Appendix A).

In addition to the working level exposures from the inhalation of short-lived radon-222 daughter products listed in table 4.2-6, radiation doses from radon-222 emissions also occur to body organs from the inhalation and ingestion of lead-210 formed from the decay of the released radon-222. Data on the doses to the population of the United States from lead-210 from radon-222 released from uranium milling operations is presented in detail in ORNL79. Our preliminary evaluation of these data indicated that the health impact resulting from the organ doses from lead-210 is smaller than the health impact from the inhalation of the short-lived radon-222 daughters. The health risk data presented in this report does not include these small additional incremental risks from lead-210.

Table 4.2-7 estimates the individual lifetime risks and number of fatal cancers to the population resulting from these doses and working level exposures. The lifetime cancer risk to the highest exposed group of individuals is estimated to be about $1x10^{-2}$ and results primarily from emissions from the tailings disposal area. The lifetime cancer risks to the average individual in the regional and United States populations are estimated to be about 10^{-5} and 10^{-8}, respectively.

The number of cancers per year of plant operation is estimated to be 0.01 to the population living in the region around the plant and 0.03 to the population of the United States. These result primarily from radon-222 emission from the tailings disposal area.

Table 4.2-5. Annual radiation doses from radioactive particulate emissions from the model uranium mill

Organ	Milling operations		Tailings disposal area		Total	
	Maximum individual (mrem/y)	Population (person-rem/y)	Maximum individual (mrem/y)	Population (person-rem/y)	Maximum individual (mrem/y)	Population (person-rem/y)
Lung	2.1E+2	1.7	1.4E+2	1.7	3.5E+2	3.4
Bone	8.4E+1	1.1	2.8E+2	2.8	3.6E+2	3.9
Kidney	2.7E+1	4.3E-1	1.3E+2	2.2	1.6E+2	2.6
Liver	1.3E+1	1.5E-1	9.7E+1	1.4	1.1E+2	1.6
Thyroid	2.0E+1	2.6E-1	1.3E+2	1.6	1.5E+2	1.9
G.I. Tract	1.0E+1	1.0E-1	8.0E+1	1.1	9.0E+1	1.2
Other soft tissue	2.2E+1	2.7E-1	1.3E+2	1.8	1.5E+2	2.1

Table 4.2-6. Working level exposures from radon-222 emissions from model uranium mill

Mill area	Maximum individual (WL)	Regional population (person-WL)	National population[a] (person-WL)
Milling operations	2.3E-4	2.2E-2	6.7E-2
Tailings disposal area	4.7E-3	5.2E-1	1.5
Total	4.9E-3	5.4E-1	1.6

[a] Calculated from data in ORNL79--Table 2.1 where a one kCi radon-222 release is estimated to result in an exposure of 8.0E+4 person-pCi/m^3 to the population of the United States. This is equivalent to an exposure to 0.56 person-working levels based on an assumption of a 70 percent equilibrium of the radon-222 daughter products (100 pCi/L radon-222 = 0.7 WL) which is considered to be representative of indoor exposure conditions (Ge78).

Table 4.2-7. Individual lifetime risks and number of fatal cancers due to radioactive emissions from the model uranium mill

Mill area	Individual lifetime risks		
	Maximum	Average individual	
	Individual	Region	United States
Uranium milling operations			
Particulates	8.9E-4	2.3E-7	--
Radon-222	3.4E-4	9.0E-7	4.5E-10
Tailings disposal area			
Particulates	1.7E-3	5.8E-7	--
Radon-222	7.0E-3	2.2E-5	1.1E-08
Total	9.9E-3	2.4E-5	1.1E-08

Mill area	Expected fatal cancers per year of operation Region (Fatal cancers)	United States (Fatal cancers)
Uranium milling operations		
Particulates	1.2E-4	--
Radon-222	4.6E-4	1.4E-3
Tailings disposal area		
Particulates	3.0E-4	--
Radon-222	1.1E-2	3.3E-2
Total	1.2E-2	3.4E-2

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EPA73 Environmental Protection Agency, 1973, Environmental Analysis of Uranium Fuel Cycle, Part I, Fuel Supply, EPA-520/9-73-003-B, Washington, D.C.

EPA76 Environmental Protection Agency, 1976, Environmental Analysis of the Uranium Fuel Cycle, Part IV, Supplementary Analysis-1976, EPA-520/4-76-017, Washington, D.C.

Ge78 George A.C., and Breslin, A.J., 1978, The Distribution of Ambient Radon and Radon Daughters in Residential Buildings in the New Jersey--New York Area, Presented at Symposium on the Natural Radiation Environment III, Houston, Texas.

GJO79 Grand Junction Office, 1979, Statistical Data of the Uranium Industry, GJO-100(79), Department of Energy, Grand Junction, Colo.

Ma78 Magno, P., 1978, Radon-222 Releases from Milling Operations, Testimony before the Atomic Safety and Licensing Board in the Matter of Perkins Nuclear Station, May 16, 1978.

NRC77a Nuclear Regulatory Commission, 1977, Final Environmental Statement Related to Operation of Bear Creek Project, NUREG-0129, Washington, D.C.

NRC77b Nuclear Regulatory Commission, 1977, Final Environmental Statement Related to Operation of Lucky McGas Hills Uranium Mill, NUREG-0357, Washington, D.C.

NRC77c Nuclear Regulatory Commission, 1977, Draft Environmental Statement Related to Operation of Moab Uranium Mill, NUREG-0341, Washington, D.C.

NRC77d Nuclear Regulatory Commission, 1977, Effluent Monitoring Requirements for Uranium Mills, Regulatory Guide 4.14, Washington, D.C.

NRC78 Nuclear Regulatory Commission, 1978, Draft Environmental Statement Related to Operation of Morton Ranch Uranium Mill, NUREG-0439, Washington, D.C.

NRC79a Nuclear Regulatory Commission, 1979, Generic Environmental Impact Statement on Uranium Milling, NUREG-0511, Washington, D.C.

REFERENCES--continued

ORNL79 Travis C.C., et al., 1979, A Radiological Assessment of Radon-222 Released from Uranium Mills and other Natural and Technologically Enhanced Sources, (NUREG/CR-0573), ORNL/NUREG-55 Oak Ridge, Tennessee.

ORNL75 Sears M.B., et al., 1975, Correlation of Radioactive Waste Treatment Costs and the Environmental Impact of Waste Effluents in the Nuclear Fuel Cycle for Use in Establishing "as Low as Practicable" Guides--Milling of Uranium Ores, ORNL-TM-4903, Vol. 1, Oak Ridge, Tennessee.

NRC79b Nuclear Regulatory Commission, 1979, Final Environmental Statement Related to Operation of the Sweetwater Uranium Project, NUREG-0403, Washington, D.C.

USFS78 U.S. Forest Service, 1978, Draft Environmental Statement for Homestakes Mining Company's Pitch Project, Washington, D.C.

4.3 Phosphate Industry

Phosphate rock is the starting material for the production of all other phosphate products. The basic operations of the phosphate industry are mining the ore which contains the phosphate rock and processing it to produce phosphoric acid and elemental phosphorus. These two products are then combined with various other chemicals to produce fertilizers, detergents, animal feeds, food products, and other phosphorus-derived chemicals. The most important use of phosphate rock is in the production of fertilizers, accounting for approximately 80 percent of the United States production of the phosphate rock.

Phosphate deposits contain appreciable quantities of natural radioactivity, principally uranium-238 and members of its decay series. Uranium concentrations in phosphate deposits range from 10 to 100 times the concentration of uranium in the natural terrestrial environment. Uranium concentrations in mined phosphate ores range up to 10 percent of the concentration in ores currently mined for the recovery of uranium. The mining and processing of phosphate ores therefore result in the atmospheric releases of radioactivity and in the production of large quantities of waste materials which also are a potential source of atmospheric emissions.

Because of the diversity of the operations in the phosphate industry which may be performed at widely separated locations, the radiological impact of the industry has been assessed by grouping the operations that are usually carried out at the same location: mining and beneficiation, ore drying and grinding, phosphoric acid production by the wet process, and elemental phosphorus production by the thermal process.

4.3A Mining and Beneficiation

4.3A.1 General Description

About 120 million metric tons of phosphate ore are mined in the United States each year. There are about 35 phosphate mines in the United States, 17 in Florida, 7 in Tennessee, 1 in North Carolina, 4 Idaho, 3 in Montana, 1 in Wyoming, and 2 in Utah (TVA 74). Florida produces the largest amount of phosphate rock; of the ore mined in 1973 about 91 percent was mined in Florida. Table 4.3A-1 lists the current phosphate mines in the Bone Valley deposit area of Florida.

Table 4.3A-1. Location and capacity of phosphate mines in Bone Valley Region, Florida, 1978 (RHS78)

Company	Mine	County	Annual Production (million tons/y)	Year Opened	Projected Closing
Agrico Chemical Co.	Payne Creek	Polk	3.0	1965	NA
	Saddle Creek	Polk	1.5	NA	1983
	Fort Green	Polk	3.0	1975	NA
	Pierce	Polk	NA	NA	NA
	Boyette	Hillsborough	NA	NA	NA
Borden, Inc.	Tenoroc	Polk	1.0	1950	1978
Brewster Phosphates	Haynesworth	Polk	3.0	1967	1990
	Lonesone	Hillsborough	2.8	1976	1996
Gardinier Inc.	U.S. Phos. Prod.	Polk	1.8	1967	1980
W.R. Grace & Co.	Bonny Lake	Polk	2.3	1945	1980
	Hooker's Prairie	Polk	2.8	1977	1994
International Minerals and Chemicals	Clear Springs	Polk	2.5	1965	1985
	Kingsford	Polk	4.5	1966	1995
	Noralyn Phosphoric	Polk	5.0	1949/1974	1990
Mobile Chemical Corp.	Ft. Meade	Polk	3.0	1967	1990
	Nichols	Polk	1.5	1972	1990
Swift Agricultural Chemical Corp.	Silver City	Polk	2.0	1964	1983
	Watson	Polk	1.0	1937	1990
USS Agri-Chemical	Rockland	Polk	2.0	1968	1986

NA Not available.

The United States production of phosphate rock will increase to about 60 million metric tons per year by 1995 when it will level off and then decline after the year 2000 (EPA78).

The uranium concentration in phosphate ore ranges up to 10 percent of the uranium concentration in the ore currently mined for uranium recovery, but because of the large amount of phosphate ore mined, the amounts of radioactivity involved, including the wastes, are approximately the same for phosphate ore processing and uranium ore processing (Gu75).

Ore used in the production of phosphoric acid usually requires beneficiation, the process where the phosphate rock is separated from the matrix by washing, screening, and flotation. This process results in large quantities of wastes consisting of equal quantities of sand tailings and slimes.

The data indicate that the radionuclides of the uranium, actinium, and probably, thorium decay series are in equilibrium and that the beneficiation process does not significantly alter this state (Gu75), although beneficiation does produce a redistribution of the radionuclide concentrations among the phosphate rock, slimes, and sand tailings. Phosphate rock, which is one-third of the mined ore, contains 40 percent of the total activity of the mined ore. The rest of the radioactivity is accounted for in the slimes (48 percent) and in the sand tailings (12 percent).

4.3A.2 Process Description

Mining

The profile of a typical phosphate deposit in the Bone Valley region of Florida is shown in figure 4.3A-1.

Almost all phosphate ore is mined by strip mining techniques. A typical mining operation (Ho76) in Florida has one or more large excavating machines called draglines which mine the ore in strips measuring 99 m wide by 10 m deep over a length of 60 m to 1000 m. During the initial box-cut the overburden is placed on virgin ground. As the mining progresses, the dragline returns the overburden into the adjacent mined-out area. The dragline casts matrix at the rate of 1100 m^3 per hour into a well that is

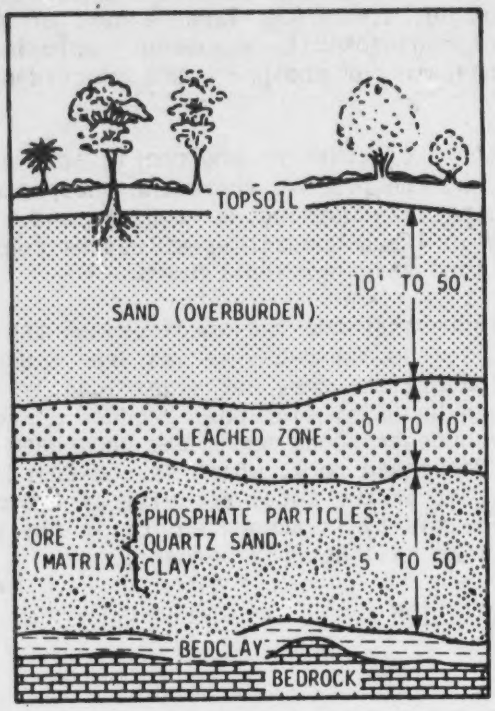
TOPSOIL
SAND (OVERBURDEN)
10' TO 50'
LEACHED ZONE
0' TO 10'
ORE (MATRIX)
PHOSPHATE PARTICLES
QUARTZ SAND
CLAY
5' TO 50'
BEDCLAY
BEDROCK

Figure 4.3A-1. Profile of a typical phosphate deposit

excavated on unmined ground. A dragline production rate of 910 cubic meters per hour is required to feed a plant producing 400 metric tons of finished product per hour. Three high pressure water guns jet water into the matrix, breaking it into a slurry that slumps to the lower part of the well where the slurry at 35-40 percent solids is then pumped to the beneficiation plant. Pumping of matrix slurry (normally over a distance of 3-10 km and in one case, 16 km) is the standard method of transporting from pit to plant in Florida.

All mines in Central Florida operate approximately 24 hours a day since they operate 20 to 21 shifts per week.

The cross section of a mining area is shown in figure 4.3A-2. Due to the topography, the surface area of a mined strip is about 12 hectares.

About 160 hectares of land are mined per year at a typical Florida mine removing 10 million cubic meters of overburden and mining 7 million cubic meters of matrix (EPA78).

Four categories of reclamation are currently in use (Ro77): two deal with the reclamation of mined areas and involve balancing the available materials (i.e., overburden and sand tailings) with proposed reclamation contours; the third method treats special problems encountered in reclaiming slime ponds; the fourth method involves mixing sands and clays. The latter method is presently used by only one company and accounted for only two parcels of reclaimed land as of 1977.

Beneficiation

In the washing and beneficiation process (Pa78), phosphate rock is separated from sand tailings and clay slimes through a series of screening and flotation steps (figure 4.3A-3).

Sand tailings are used for dam construction, slime ponds, or land reclamation. Slime disposal involves large settling areas (30 to 60 percent of mined lands), and reclamation to support agricultural uses requires 5-20 years (EPA78, Ro77).

4.3A.3 Emissions of Radionuclides

The only significant radioactive emission from the mining and beneficiation operations is the release of radon-222. Particulate emissions do not seem to be a problem. No measurement studies have been made of the actual releases of radon-222 during these operations. So for this report it will be assumed that all the available (20 percent of the total) radon-222 in the ore is released, one-half of the radon-222 being released during the mining operation and one-half being released during the beneficiation process.

4.3A.4 Model Facility

This preliminary assessment of the phosphate industry estimates the radioactive emissions and the consequent health

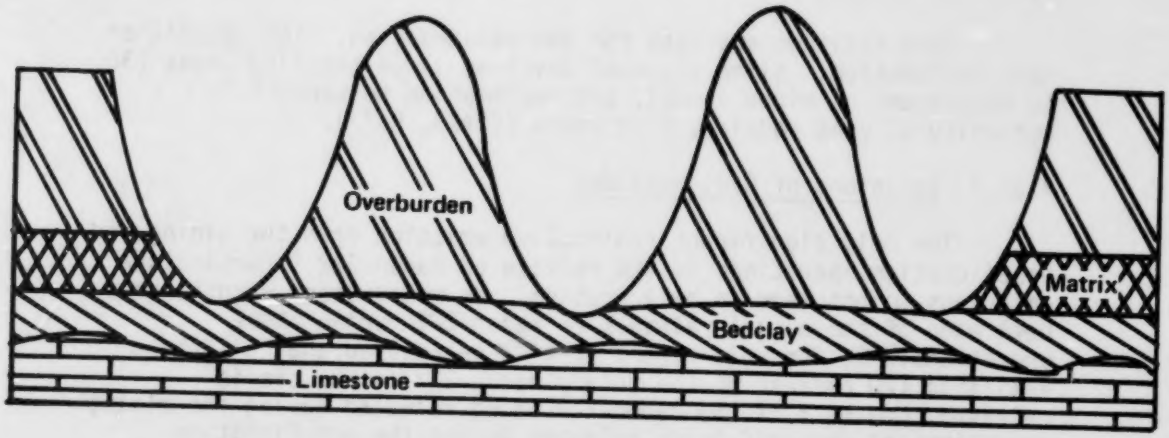
Overburden
Matrix
Bedclay
Limestone

Figure 4.3A-2. Cross section of a typical mined-out strip of a phosphate mine (Ro77)

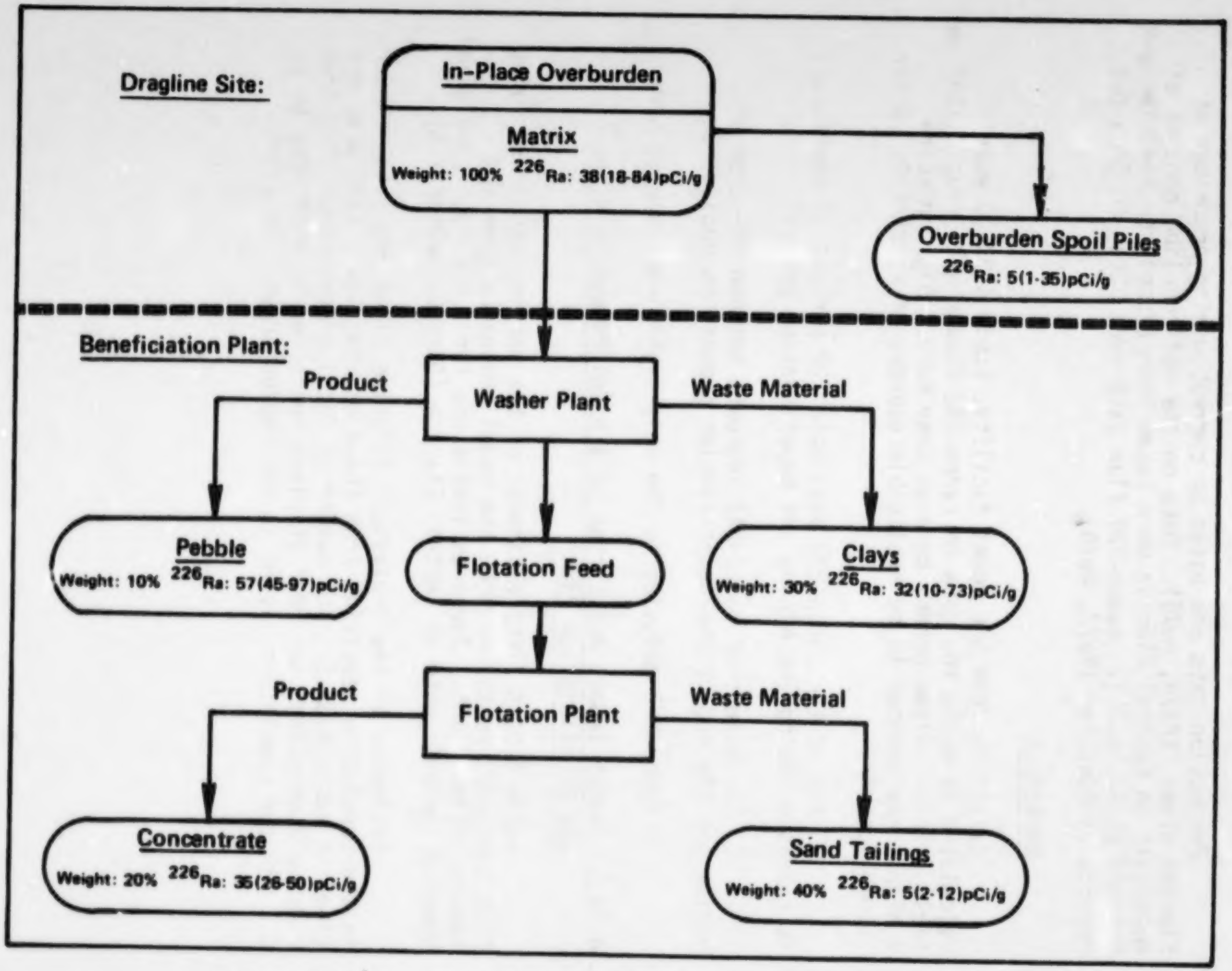
Dragline Site:
In-Place Overburden
Matrix
Weight: 100% ^{226}Ra: 38(18-84)pCi/g
Overburden Spoil Piles
^{226}Ra: 5(1-35)pCi/g
Beneficiation Plant:
Washer Plant
Product
Waste Material
Pebble
Weight: 10% ^{226}Ra: 57(45-97)pCi/g
Flotation Feed
Clays
Weight: 30% ^{226}Ra: 32(10-73)pCi/g
Flotation Plant
Product
Waste Material
Concentrate
Weight: 20% ^{226}Ra: 35(26-50)pCi/g
Sand Tailings
Weight: 40% ^{226}Ra: 5(2-12)pCi/g

Figure 4.3A-3. Flow diagram for major products and mineral waste products of phosphate rock mining and beneficiation (Ro78)

impacts by developing model facilities and assigning appropriate values for parameters which affect releases of radioactivity to the atmosphere at each model facility (table 4.3A-2).

Production data are based on current mining practices at Florida mines (EPA78, Ho76). Data on the radionuclide content of materials in Central Florida were taken from reports by Roessler and Guimond (Ro77, Gu75). Radon-222 flux data were obtained from two reports by Roessler (Ro77, Ro78).

Emissions

Emissions from the model facility, table 4.3A-3, were calculated by using the data on radon-222 fluxes and areas listed in table 4.3A-2. Slime ponds, because they have a high moisture content, are assumed to be negligible sources of radon-222. Other assumptions are:

1) All of the radon-222 available (20 percent of the total) is released during the mining and beneficiation process.

2) The amount of radon-222 ingrowth between the time of mining and the end of the beneficiation process is negligible.

3) Radon-222 emission is the only significant source term.

4.3A.5 Health Impact Assessment of a Model Phosphate Mine and Beneficiation Plant

Table 4.3A-4 lists estimates of the working level exposures from radon-222 emissions from the model phosphate mine and beneficiation plant. These estimates are for a moderate population density, generic site in Central Florida (Site C, Appendix A).

Estimates of the individual lifetime risks and fatal cancers in the population resulting from these working level exposures are given in table 4.3A-5. The number of fatal cancers resulting from radon-222 emissions per year of plant operation is estimated to be 0.1 for the population living in the region around the plant.

Table 4.3A-2. Model phosphate mine and beneficiation plant

Parameter	Value
Total area of mine	4000 ha
Active mining area	480 ha
Undisturbed area	2000 ha
Reclaimed land	520 ha
Slime pond area	1000 ha
Annual production	
Ore	9.0E+6 t 7.0E+6 m^3
Phosphate rock	3.0E+6 t
Overburden removed	1.4E+7 t
Leached zone material removed	5.0E+6 t
Mine lifetime	24 years
Production year[a]	12th year
Radionuclide content[b]--(pCi/g)	
Overburden (3 m depth)	
Uranium-238	3.0
Thorium-232[c]	3.0E-2
Leached zone material (1 m depth)	
Uranium-238	1.9E+1
Thorium-232	1.0E-1
Ore (4 m seam)	
Uranium-238	3.8E+1
Thorium-232[c]	3.0E-1
Radon-222 flux for land types (pCi/m^2-s):	
Undisturbed area	3.0E-1
Tailings	1.6
Overburden	4.4

---

[a] Production and emission data are based on the 12th year of plant operations.

[b] Members of the ^{238}U and ^{232}Th decay series are in equilibrium.

[c] The estimate is based on measured ratio of uranium-238/thorium-232 in phosphate rock

Table 4.3A-3. Radon-222 emissions from the model phosphate mine and beneficiation plant

Source	Radon-222 (Ci/y)
Ore removal and beneficiation	7.0E+1
Overburden removal	8.0
Leached zone material removed	1.9E+1
Mined-out area (before reclamation)	5.9E+2
Reclaimed land	6.4E+2
Total	1.3E+3

Table 4.3A-4. Working level exposures from radon-222 emissions from the model phosphate mine and beneficiation plant

Source	Maximum individual (WL)	Regional population (person-WL)
Mine and beneficiation plant	2.2E-4	4.9

Table 4.3A-5. Individual lifetime risks and the number of fatal cancers due to radioactive emissions from the model phosphate mine and beneficiation plant

	Individual lifetime risks		Expected fatal cancers per year of operation
Source	Maximum individual	Average individual	(Fatal cancers)
Mine and beneficiation plant	3.3E-4	4.7E-6	1.06-1

4.3B Phosphate Ore Drying and Grinding Facility

4.3B.1 General Description

After phosphate rock has been beneficiated, it is usually dried and ground to a uniform particle size to process it efficiently. Ore drying and grinding operations release significant amounts of particulate material (phosphate rock dust) and are a source of radon-222 emissions to the atmosphere. Dry grinding is currently the general practice, but wet grinding methods, which would reduce dust releases, are being studied.

There are approximately 20 ore drying and grinding facilities in the United States which process about 50 million metric tons of phosphate rock annually. A typical facility processes about 2 to 3 million metric tons of rock each year. They are either separate facilities or associated with phosphoric acid or elemental phosphorus plants.

The growth of the ore drying operation keeps pace with phosphate mining operations. The industry is expected to grow from about 50 million metric tons per year of phosphate rock in the 1970's to about 60 million metric tons per year through the year 2010 (EPA78).

4.3B.2 Process Description

Following the washing process the phosphate rock is transferred to the drying and storage area where the wet rock is dried in large rotating drums or on a fluid bed. Wet phosphate rock entering the ore drying operation weighs 1.4 g/cc with a 6 to 10 percent moisture content and is dried to a 2 to 3 percent moisture content (Ho76b). After drying the rock is separated according to size and grade, and then stored. In some cases, material from the dryers is ground using ball mills before storage.

The predominant airborne emissions are fine rock dust from drying and grinding operations. Phosphate rock dryers are usually equipped with dry cyclones followed by a fine particulate collector. Control devices, such as baghouse collectors (Pa78), electrostatic precipitators, and wet-type collectors (Sp67), indicate 98 plus percent collection efficiency.

4.3B.3 Emissions of Radionuclides

Phosphate rock dust is a source of particulate radioactivity in the atmosphere because the dust particles have approximately the same specific activity (pCi/g) of uranium-238 and its decay products as the phosphate rock (Pa78). The radioactivity of the dust particles varies according to the type of phosphate ore mined. In Central Florida the concentration of radium-226 in phosphate rock ranges from 26 to 97 pCi/g (Ro78). The amount of particulate radioactivity released per year may be calculated by multiplying the amount of rock dust released per year by the concentrations of the radionuclides in the phosphate rock. For example, if a facility releases 1 x 10^8 grams of rock dust per year and the concentration of radium-226 in phosphate rock is 40 pCi/g, the facility then released approximately 4 mCi/y of radium-226 to the atmosphere.

Since there are no data on the amount of radon-222 released during the drying and grinding operations, it has been assumed that radon-222 is in equilibrium with radium-226 and that all the available (20 percent of the total) radon-222 is released during the drying process.

4.3B.4 Model Facility

In order to estimate radioactive emissions and health impacts from ore drying and grinding operations, a model facility was developed by assigning the various parameters that are important in assessing impacts (table 4.3B-1). These parameters were based on reports by Partridge (Pa78) and Guimond (Gu75).

Emissions

The atmospheric emissions of radionuclides from the model ore drying and grinding facility are listed in table 4.3B-2. The particulate emission rates were based on the amount of rock dust released per year and the radionuclide concentrations measured in the dust effluents (Pa78). The radon-222 emission rate was estimated from the amount of phosphate rock processed per year, the radium concentration in the rock, and the assumption that the radon-222 equilibrium with radium-226 was reestablished after beneficiation and that all the available radon-222 (20 percent of the total) was released during the operations.

4.3B.5 Health Impact Assessment of a Model Phosphate Ore Drying and Grinding Facility

The estimated annual radiation doses and working level exposures resulting from radioactive emissions from the model ore drying and grinding facility are listed in tables 4.3B-3 and

4.3B-4. These estimates are made for a moderate population density, generic site in Central Florida (Site C, Appendix A).

Estimates of the individual lifetime risks and the number of fatal cancers in the population resulting from these doses and working level exposures are given in table 4.3B-5. The lifetime cancer risk to the highest exposed group of individuals is estimated to be about 5×10^{-4} and results primarily from particulate emissions from the facility. The lifetime cancer risk to the average individual in the region is estimated to be about 2×10^{-7}.

The number of fatal cancers per year of plant operation is estimated to be 3.6×10^{-3} to the population living in the region around the plant. Particulate releases and radon-222 emissions from the facility contribute about equally to this number.

Table 4.3B-1. Model phosphate ore drying and grinding facility

Parameter	Value
Annual process rate (wet rock)	2.7E+6 t
Moisture Content	
Wet rock	8%
Dry rock	3%
Plant lifetime	20 years
Plant stacks	
Stack height	10 meters
Stack diameter	1 meter
Particulate releases	1.1E+8 g/y
Radionuclide Content--(pCi/g)[a]	
Phosphate rock	
Uranium-238	4.2E+1
Thorium-232	4.4E-1

[a] Members of the ^{238}U and ^{232}Th decay series are in equilibrium.

Table 4.3B-2. Emissions of radionuclides from the model phosphate ore drying and grinding facility

Source	Emissions (Ci/y)
Particulates	
^{238}U decay series (each member)	5.0E-3
^{232}Th decay series (each member)	2.0E-4
Radon-222	2.0E+1

Table 4.3B-3. Annual radiation doses from radioactive particulate emissions from the model ore drying and grinding facility

Organ	Maximum individual (mrem/y)	Population (person-rem/y)
Lung	5.4E+1	1.7E+1
Bone	7.9E+1	1.8E+1
Kidney	2.7E+1	9.5
Liver	1.5E+1	5.0
Thyroid	2.5E+1	7.4
G.I. tract	1.0E+1	3.5
Other soft tissue	2.5E+1	7.5

Table 4.3B-4. Working level exposures from radon-222 emissions from the model ore drying and grinding facility

	Maximum individual (WL)	Regional population (person-WL)
Ore drying and grinding	5.3E-5	8.4E-2

Table 4.3B-5. Individual lifetime risks and the number of fatal cancers due to radioactive emissions from the model ore drying and grinding facility

	Individual lifetime risks		Expected fatal cancers per year of operation
Source	Maximum individual	Average individual	(Fatal cancers)
Particulates	4.5E-4	8.5E-8	1.8E-3
Radon-222	8.0E-5	8.5E-8	1.8E-3
Total	5.3E-4	1.7E-7	3.6E-3

4.3C Phosphoric Acid Plant

4.3C.1 General Description

Phosphoric acid, used mainly in the manufacture of high-analysis fertilizers, is produced by the wet process method. At a typical phosphate plant complex, the phosphoric acid is the starting material for ammonium phosphate and triple superphosphate fertilizers which are manufactured in other units of the complex.

The total phosphoric acid production in the United States was about 9 million metric tons of the acid (as phosphorus pentoxide) in 1976, produced from more than 50 million metric tons of phosphate rock. The waste generated, in the form of gypsum, is five times the weight of the phosphoric acid (as phosphorus pentoxide) produced. This means that in producing 9 million tons of phosphoric acid, 45 million metric tons of gypsum per year are produced(EPA74). The disposal area for gypsum required per year is approximately 1200 cubic meters for each daily metric ton of phosphorus pentoxide produced. A large plant produces 1,000 metric tons of acid (as phosphorus pentoxide) per day. Therefore, 35 plants would require 35 million cubic meters of storage area per year for the gypsum produced (S168).

4.3C.2 Process Description

Phosphoric acid, which is usually manufactured by the wet process method, is the basic building block from which essentially all mixed fertilizer used in the United States is made. The raw materials used in this process are ground phosphate rock, 93 percent sulfuric acid and water. Phosphate rock is mixed with sulfuric acid after the acid has first been diluted with water to a 55 to 70 percent sulfuric acid concentration. A diagram of the process is shown in figure 4.3C-1. The simplified overall reaction is represented by the following equation:

$$3Ca_3(PO_4)_2 + 9H_2SO_4 + 18H_2O$$

$$= 6H_3PO_4 + 9CaSO_4 \cdot 2H_2O \qquad (1)$$

Phosphate rock is not the pure compound indicated above, but a fluoroappitite material containing minor quantities of fluorine, iron, aluminum, silica and uranium. Following the reaction in the digester, the mixture of phosphoric acid and gypsum is pumped to a filter

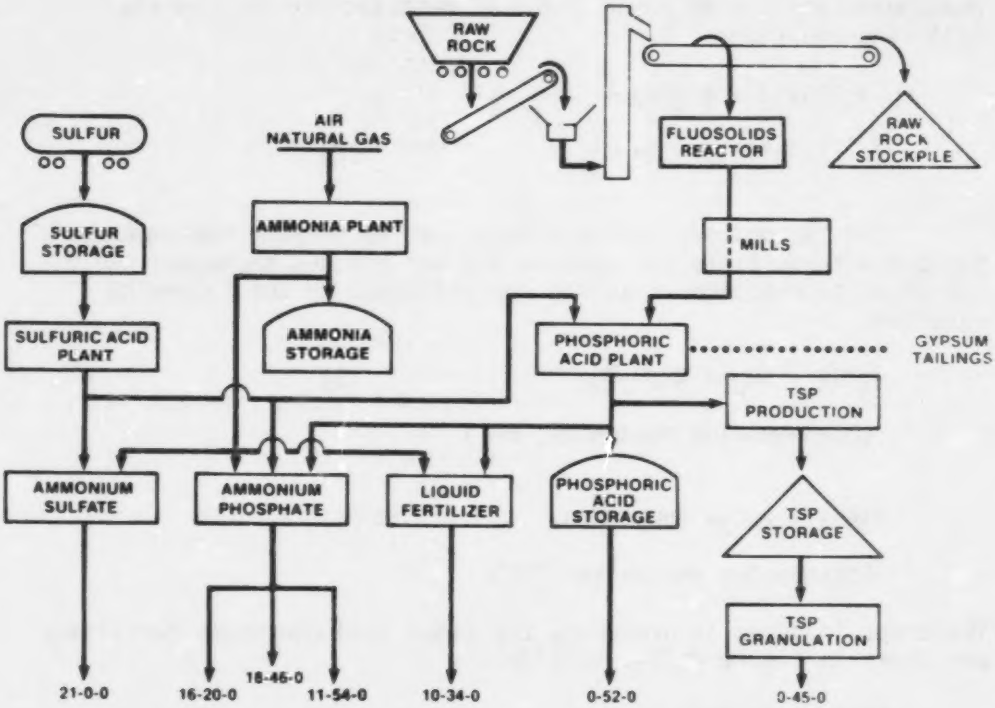
RAW ROCK
SULFUR
AIR
NATURAL GAS
FLUOSOLIDS REACTOR
RAW ROCK STOCKPILE
SULFUR STORAGE
AMMONIA PLANT
MILLS
SULFURIC ACID PLANT
AMMONIA STORAGE
PHOSPHORIC ACID PLANT
GYPSUM TAILINGS
TSP PRODUCTION
AMMONIUM SULFATE
AMMONIUM PHOSPHATE
LIQUID FERTILIZER
PHOSPHORIC ACID STORAGE
TSP STORAGE
TSP GRANULATION
21-0-0
16-20-0
18-46-0
11-54-0
10-34-0
0-52-0
0-45-0

Figure 4.3C-1. Flow diagram of the wet process

which mechanically separates the particulate gypsum from the phosphoric acid (approximately 30 percent phosphorus pentoxide concentration). An enormous amount of the byproduct gypsum is produced--each metric ton of phosphorus pentoxide, as phosphoric acid, produces approximately 5 metric tons of gypsum. Normally the gypsum is sluiced with process water from the plant to the disposal area. The phosphoric acid separated from the gypsum is collected for further processing (EPA74).

Fertilizer Production

The next step in the process (figure 4.3C-1) is the production of fertilizer, usually triple superphosphate. The raw materials used in its manufacture are ground phosphate rock and phosphoric acid. The basic chemical reaction is shown by the following equation:

$$\begin{aligned} &Ca_3(PO_4)_2 + 4H_3PO_4 + 3H_2O \\ &= 3Ca(H_2PO_4)_2 \cdot 2H_2O \qquad (2) \end{aligned}$$

The two primary raw materials used to produce the ammonium phosphate fertilizers are ammonia and wet process phosphoric acid. The chemical reactions involved are indicated by the following equations:

$$H_3PO_4 + NH_3 = NH_4H_2PO_4 \qquad (3)$$

(monoammonium phosphate, MAP)

$$H_3PO_4 + 2NH_3 = (NH_4)_2HPO_4 \qquad (4)$$

(diammonium phosphate, DAP)

The steps involved in producing the ammoniated phosphate fertilizer are shown in figure 4.3C-1 (EPA74a).

Control Technology

Wet scrubbers are used on all plant stacks to control process effluents and dust emissions from ore and product handling areas. Radon-222 emissions from plant operations and gypsum piles are not controlled.

4.3C.3 Emissions of Radionuclides

Atmospheric emissions from the manufacture of phosphoric acid are phosphate rock dust from ore handling operations, particulates of the ammonium phosphate and triple superphosphate products, and radon-222. In addition, radon-222 is released in significant amounts from the gypsum pile, since most of the radium-226 contained in the phosphate rock is found in the gypsum byproduct.

The radionuclide equilibrium that exists in the phosphate rock is disrupted during the chemical process with approximately one percent of radium-226, 60 to 80 percent of thorium-230, and 80 percent of the uranium being dissolved in the acid (Gu75). The dissolved radionuclides are accounted for finally in the fertilizer products.

Some wet process plants (in Western United States) include a calcining process to remove organic material from the phosphate rock. Therefore, this is an additional source of particulates which include polonium-210 and lead-210 (ORP78).

4.3C.4 Model Facility

The parameters of a model facility needed to estimate the emissions and the health impacts from phosphoric acid production are listed in table 4.3C-1. Production data and most other parameters are based on reports on wet process plants in Florida (Pa78) and in Idaho (ORP78). The exhalation rate of radon-222 from a gypsum pile was obtained from a draft report by Horton (Ho79).

Emissions

Emissions from the model facility are listed in table 4.3C-2. Particulate emissions are based on a report by Partridge (Pa78). Radon-222 emissions from the gypsum pile were determined from the exhalation rate and area given in table 4.3C-1. The radon-222 emissions from the plant are based on the amount of ore processed per year and the assumption that all the radon-222 in equilibrium with radium-226 contained in the ore is released during the chemical process.

Table 4.3C-1. Model phosphoric acid plant

Parameter	Value
Annual process rate	2.0E+9 kg
Annual production[a]	
Phosphoric acid	4.3E+8 kg
Diammonium phosphate	2.0E+8 kg
Triple superphosphate (TSP)	9.1E+7 kg
Plant lifetime	20 years
Plant stacks	
Height	10 meters
Diameter	1 meter
Flow rate	2400 m^3/min
Gypsum pile	
Effective emission height	10 meters
Area	4.9E+5 m^2
Amount stored	2.0E+9 kg/y
Radon-222 exhalation rate	1.6E+3 pCi/m^2-min
Annual particulate releases (grams)	
Diammonium phosphate (DAP)	1.5E+8
Triple superphosphate (TSP)	1.4E+8
Product storage area (DAP+TSP)	2.1E+8
Phosphate rock	5.1E+7
Radionuclide content--(pCi/g)	
Phosphate rock[b]	
Uranium-238	4.2E+1
Thorium-232	4.4E-1
Phosphoric acid (52%)	
Radium-226	8.4E+2
Uranium-238	5.1E+4
Triple superphosphate	
Radium-226	2.1E+1
Uranium-238	5.8E+1
Uranium-234	5.8E+1
Thorium-230	4.8E+1
Polonium-210	3.2E+1

See footnotes at end of table.

Table 4.3C-1. Model phosphoric acid plant--continued

Parameter	Value
Radionuclide content--(pCi/g)-continued	
Ammonium phosphate	
Radium-226	5.6
Uranium-238	6.3E+1
Uranium-234	6.3E+1
Thorium-230	6.5E+1
Thorium-232	4.0E-1
Thorium-238	8.0E-1
Gypsum	
Radium-226	3.3E+1
Uranium-238	6.0
Uranium-234	6.2
Thorium-230	1.3E+1
Thorium-232	2.7E-1
Thorium-238	1.4

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[a] Weights given are expressed as weight of P_2O_5.

[b] Members of the ^{238}U and ^{232}Th decay series are in equilibrium.

Table 4.3C-2. Emissions of radionuclides from the model phosphoric acid plant

Source	Emissions (Ci/y)
Particulates	
Uranium-238	2.3E-2
Uranium-234	2.3E-2
Thorium-228	4.8E-4
Thorium-230	2.0E-2
Thorium-232	4.8E-4
Radium-226	1.1E-2
Polonium-210	[a]1.2E-2
Lead-210	[b]1.2E-2
Radium-228	4.8E-4
Radon-222	
Plant operations	8.0E+1
Gypsum pile	4.0E+2

[a] Emission rate is calculated from particulate releases and the concentration of ^{210}Po measured in TSP and phosphate rock.

[b] Measurements of ^{210}Pb were not made; but releases, as a first approximation, are assumed to be at the same rate as ^{210}Po.

4.3C.5 Health Impact Assessment of a Model Phosphoric Acid Plant

The estimated annual doses and working level exposures resulting from radioactive emissions from the model phosphoric acid plant are listed in tables 4.3C-3 and 4.3C-4. These estimates are for a moderate population density around a generic site in central Florida (Site C, Appendix A). The highest exposed group of individuals is assumed to be located 700 m from the center of the model plant.

Table 4.3C-5 lists the estimates of the individual lifetime risks and fatal cancers in the population resulting from these doses and working level exposures. The lifetime cancer risk to the highest exposed group of individuals is estimated to be about 1.6 x 10^{-3}. This number results from the release of particulates and radon-222 from the plant and radon-222 emissions from the gypsum pile. The lifetime cancer risk to the average individual in the region is estimated to be about 2 x 10^{-6}. The number of fatal cancers per year of plant operation is estimated to be about 0.05 to the population living in the region around the plant. This number is primarily due to radon-222 emissions from the gypsum pile.

Table 4.3C-3. Annual radiation doses from radioactive particulate emissions from the model phosphoric acid plant

Organ	Maximum individual (mrem)	Population (person-rem)
Lung	85	46
Bone	110	45
Kidney	37	21
Liver	19	11
Thyroid	30	16
G.I. tract	13	7.8
Other soft tissue	31	16

Table 4.3C-4. Working level exposures from radon-222 emissions from the model phosphoric acid plant

	Maximum individual (WL)	Regional population (person-WL)
Plant operations	1.3E-4	3.4E-1
Gypsum pile	5.3E-4	1.7
Total	6.6E-4	2.0

Table 4.3C-5. Individual lifetime risks and number of fatal cancers due to radioactive emissions from the model phosphoric acid plant

Source	Individual lifetime risks Maximum individual	 Average individual	Expected fatal cancers per year of operation (Fatal cancers)
Phosphoric acid plant			
Particulates	6.4E-4	2.1E-7	4.5E-3
Radon-222	2.0E-4	3.3E-7	7.1E-3
Gypsum pile			
Radon-222	8.0E-4	1.7E-6	3.6E-2
Total	1.6E-3	2.3E-6	4.8E-2

4.3D Elemental Phosphorus Plant

4.3D.1 General Description

Elemental phosphorus is the starting material for the nonfertilizer part of the phosphorus industry. Over 87 percent of the elemental phosphorus produced is used in the manufacture of the high grade phosphoric acid by the furnace or "dry" process. The rest of the elemental phosphorus is either used directly or converted to other chemicals for use by the organic chemicals industry (EPA74b).

There are 9 operating elemental phosphorus plants in the United States located in Tennessee (3), Alabama (1), Idaho (2), Montana (1), and Florida (2).

Approximately 500 thousand metric tons of elemental phosphorus are produced each year (EPA76) from 5 million metric tons of phosphate rock. About 4.4 million metric tons of slag, a waste product, are also produced. Slag contains most of the radioactivity originally present in phosphate rock.

4.3D.2 Process Description

The phosphate rock is crushed, screened, and usually briquetted; the rock is then fed into calciners where it is heated to 1300°C. Calcining serves two purposes: (1) to burn out organic material and (2) to heat-harden agglomerates so they will withstand further processing steps without disintegrating. The calcined nodules pass through a proportioning building where sized coke and silica are blended into the material now called the "burden" which then moves into the electric furnace. The high temperature reaction in the furnace (1400°C--4400°C) drives off two gases, phosphorus and carbon monoxide. It leaves two molten residues, slag, and ferrophosphorus (a mixture of iron, of vanadium, chromium, etc.). A simplified chemical equation for the electric furnace reaction is:

$$\begin{aligned} &2Ca_3(PO_4)_2 + 6SiO_2 + 10C \\ &= P_4 + 10CO + 6CaSiO_3 \qquad (5) \end{aligned}$$

Off-gases are treated in electrostatic precipitators for dust removal, then in a waterspray cooler where the gaseous elemental phosphorus is condensed, collected in a sump, and pumped to

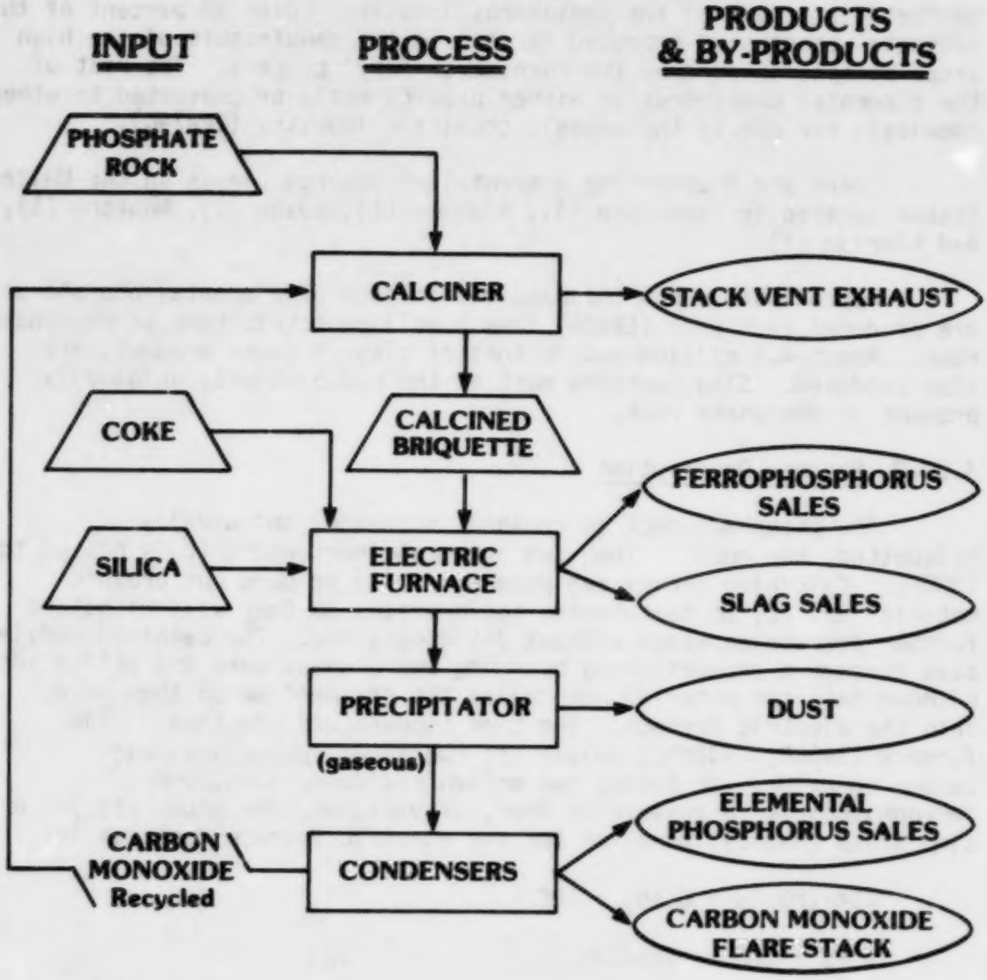
INPUT
PROCESS
PRODUCTS & BY-PRODUCTS
PHOSPHATE ROCK
CALCINER
STACK VENT EXHAUST
COKE
CALCINED BRIQUETTE
FERROPHOSPHORUS SALES
SILICA
ELECTRIC FURNACE
SLAG SALES
PRECIPITATOR
DUST
(gaseous)
ELEMENTAL PHOSPHORUS SALES
CARBON MONOXIDE Recycled
CONDENSERS
CARBON MONOXIDE FLARE STACK

Figure 4.3D-1. Flow diagram of the thermal process for production of elemental phosphorus

storage. Most of the carbon monoxide is used in the calciner but some is burned to become carbon dioxide, which is released to the atmosphere via the flare stack. A flow diagram of the process is shown in figure 4.3D-1.

Control Technology

Wet scrubbers are generally used at elemental phosphorus plants to control the amount of particulate material and soluble gases released from plant stacks. There are no controls applied to the emissions from slag material.

4.3D.3 Emissions of Radionuclides

Particulates and radon-222 are released to the atmosphere during plant operations. Because most of the radioactivity from the phosphate rock is transferred to the slag, this waste product is a source of radon-222. Slag is a molten material as it comes the furnace and, when crushed after solidification, breaks into chunks rather than small particles. Thus, the resuspension of particles in the air is probably not a problem.

Phosphate rock is subjected to a high temperature (1300°C) in the calciner which causes polonium-210 and lead-210 to volatilize. These radionuclides contribute a significant portion of the radioactivity being emitted from the elemental phosphorus plants.

The only source of data on atmospheric emissions of radioactivity from an elemental phosphorus plant is an EPA study (ORP77). However, this study did not include estimates of radon-222 emissions from the the plant or slag.

4.3D.4 Model Facility

Various parameters were assigned to a model elemental phosphorus plant (table 4.3D-1) to estimate the emissions of radioactivity to the atmosphere and the consequent health impacts. The information in this table was obtained from general phosphate industry data (EPA74b) and a report of a study at an elemental phosphorus plant (ORP77). The estimate of the area of the slag pile was based on the amount of slag produced per year, the assumption that the plant is 10 years old, and that the height of the pile is 10 meters.

The exhalation rate of radon-222 from the slag pile is a rough estimate based on a calculation (Section C.2, Appendix C) which assumes: an emanation power of 0.1 of radon-222, a radium-226 concentration of 32 pCi/g, the fraction of void spaces is 0.4, and the radon-222 diffusion coefficient is 0.02 cm^2/s.

Emissions

The atmospheric emissions of radioactive material from the model facility are listed in table 4.3D-2. The data on particulate emissions were taken from a report on an elemental phosphorus plant (ORP77). The most significant amount of particulate radioactivity released annually from the plant is 7.4 curies of polonium-210 which amount to approximately 20 percent of the polonium-210 entering the process (ORP77). Estimates of radon-222 emissions from the slag pile were based on the exhalation rate and the slag pile area listed in table 4.3D-1.

The amount of radon-222 released during the plant operations was estimated by assuming that the radon-222 is in equilibrium with the radium-226 in the phosphate rock and that all the radon-222 was released during the chemical process. Then, knowing the amount of phosphate rock processed per year and the concentration of radium-226 in the rock, a rough estimate of the amount of radon-222 released was made.

4.3D.5 Health Impact Assessment of the Model Elemental Phosphorus Plant

The estimated annual radiation doses and working level exposures that result from radioactive emissions from the model elemental phosphorus plant are listed in tables 4.3D-3 and 4.3D-4. These estimates are for a moderate population density generic site in Central Florida (Site C, Appendix A). The highest exposed group of individuals is assumed to be located 1000 m from the center of the model plant.

Estimates of the individual lifetime risks and fatal cancers to the population resulting from these doses and working level exposures are shown in table 4.3D-5. The lifetime cancer risk to the highest exposed group of individuals is estimated to be about 6 $x 10^{-3}$ and results primarily from emissions of polonium-210 from the calciner stacks at the plant. The lifetime cancer risk to the average individual in the region is estimated to be $5 x 10^{-6}$.

The number of fatal cancers per year of plant operation is estimated to be about 0.1, primarily due to emissions of polonium-210 from the plant, but the contribution of radon-222 from the slag pile is also significant.

Table 4.3D-1. Model elemental phosphorus plant

Parameter	Value
Annual process rate	
Phosphate	1.6E+9 kg
Coke	1.7E+8 kg
Silica	1.1E+8 kg
Annual production	
Elemental Phosphorus	1.1E+8 kg
Slag	1.6E+9 kg
Ferrophosphorus	2.0E+7 kg
Fluid-bed prills	1.6E+7 kg
Plant lifetime	20 years
Plant stacks	
Number of stacks	9
Height	10 meters
Diameter	1.9 meters
Flow rate	3.0E+3 m^3/min
Slag pile	
Height	10 meters
Area	100 hectares
Amount stored	1.6E+9 kg/y
Radon-222 exhalation rate	15 pCi/m^2-s
Radionuclide content--(pCi/g)	
Raw materials	
Phosphate rock[a]	26 pCi/g ^{238}U
Coke	1 pCi/g ^{238}U
Silica	1 pCi/g ^{238}U
Products and wastes	
Slag	32 pCi/g ^{226}Ra
Elemental phosphorus	0.02 pCi/g ^{226}Ra 0.2 pCi/g ^{210}Po

[a] Members of the uranium-238 decay series are in equilibrium.

Table 4.3D-2. Emissions of radionuclides from the model elemental phosphorus plant

Source	Emissions (Ci/y)
Particulates	
Uranium-238	4.0E-2
Uranium-234	4.0E-2
Thorium-230	4.0E-2
Radium-226	7.0E-3
Lead-210	2.0E-2
Polonium-210	7.4
Thorium-232	1.0E-3
Radium-228	1.0E-3
Thorium-228	1.0E-3
Radon-222	
Plant stacks	4.2E+1
Slag pile	4.5E+2

Table 4.3D-3. Annual radiation doses from radioactive particulate emissions from model elemental phosphorus plant

Organ	Maximum individual (mrem/y)	Population (person-rem/y)
Lung	740	770
Bone	570	440
Kidney	1800	1400
Liver	320	260
Thyroid	120	99
G.I. tract	30	25
Other soft tissue	120	99

Table 4.3D-4. Working level exposures from radon-222 emissions from model elemental phosphorus plant

	Maximum individual (WL)	Regional population (person-WL)
Plant operations	3.4E-5	1.7E-1
Slag pile	3.2E-4	1.8
Total	3.5E-4	2.0

Table 4.3D-5. Individual lifetime risks and the number of fatal cancers due to radioactive emissions from the model elemental phosphorus plant

Source	Individual lifetime risks Maximum individual	 Average individual	Expected fatal cancers per year of operation (Fatal cancers)
Elemental phosphorus plant			
Particulates	5.2E-3	3.1E-6	6.6E-2
Radon-222	5.1E-5	1.7E-7	3.6E-3
Slag pile			
Radon-222	4.8E-4	1.8E-6	3.8E-2
Total	5.8E-3	5.1E-6	1.1E-1

4.3E.1 Summary

Phosphate ore deposits contain a significant amount of uranium and its decay products. Because of the large amount of phosphate ore mined each year, the amount of uranium removed along with the phosphate ore roughly equals that mined each year in the uranium mining industry.

Phosphate ore processsing results in emissions of radioactive particulates and radon-222 from the processing plants and waste piles. Table 4.3E-1 summarizes the lifetime risks and the number of fatal cancers resulting from radioactive emissions from the model phosphate facilities.

In this preliminary assessment of the phosphate industry, the most significant particulate emissions come from the model elemental phosphorus plant which releases polonium-210 from the calcining operation. The release of about 7 curies per year of this radionuclide is primarily responsible for the estimated lifetime cancer risk of 6 x 10^{-3} to the highest exposed group of individuals and the fatal cancer rate of 0.1 per year in the population living around the plant. Emissions of radon-222 from the mining of phosphate ore are also significant--resulting in an estimated 0.1 fatal cancers per year in the population around the mine.

Table 4.3E-1. Summary of individual lifetime risks and number of fatal cancers due to radioactive emissions from the model facilities of the phosphate industry

Source	Individual lifetime risks		Expected fatal cancers per year of operation (Fatal cancers)
	Maximum individual	Average individual	
Mine and beneficiation plant	3.3E-4	4.7E-6	1.0E-1
Ore drying and grinding plant	5.3E-4	1.7E-7	3.6E-3
Phosphoric acid plant	1.6E-3	2.3E-6	4.8E-2
Elemental phosphorus plant	5.8E-3	5.1E-6	1.1E-1

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4.4 Steam-electric coal-fired generating stations

4.4.1 General Description

Coal is black, soft rock containing at least 50 weight percent carbon. It also contains sulfur, iron, moisture, and trace quantitities of naturally occurring radioactive materials. All radionuclides of each of the three natural series and potassium-40 are found in coal. When coal is burned, the mineral content of coal is converted to ash and slag. These waste products contain the radionuclides originally present in the coal. A fraction of the ash is released to the atmosphere; the quantity released depends upon the efficiency of the particulate control system, mineral matter content of coal, furnace design, and applicable emission control standards. Retained ash may be stored on the station site. These waste piles are sources of fugitive emissions.

Half of our coal is mined west of the Mississippi River. Kentucky is the leading coal producing State while Ohio is the leading coal consuming State.

In 1977 there were 1200 coal-fired units at 395 sites in the United States. The total installed electric generating capacity of these units was 226 GW or about 42 percent of the nation's capacity. Most of these coal-fired units are relatively small--about 50 percent (600) have a generating capacity of less than 100 MW. The remaining units range from 100 MW to 1.0 GW with about one percent exceeding 1.0 GW capacity. During 1976 electric utilities burned 480 M metric tons of coal containing about 290 curies of uranium-238 and 250 curies of thorium-232.

At least 230 large units are projected to come on line in the interval 1980 to 2001. Many of the older and smaller currently operating units will be retired during this period. The Department of Energy estimates more than 500 GW of coal-fired capacity will be on line by 2000. Table 4.4-1 presents the estimated distribution of new stations.

In addition to coal-fired utility boilers, there are 24,000 fossil-fueled industrial boilers at over 3,000 sites, 750 oil-fired utility boilers, 975 gas-fired utility boilers, 950 combustion turbines, 285 refineries, 625 steel mill furnaces, and over 13,000 coal-fed coke ovens.

Table 4.4-1. Proposed new coal fired power plants, 1979-2001[a]

Fuel	Number of units[b]	Total capacity (GW)[c]
Eastern Bituminous	95	52
Western Bituminous	97	53
Lignite	18	10
Total	210	115

[a] Source: Ri78.

[b] The number of units burning Eastern and Western bituminous coal is based on an arbitrary but quasi-equal division among the total number of units burning this rank of coal.

[c] The capacity estimate is based on 550 MW per unit.

4.4.2 Process Description

Combustion

The technology for producing electricity from the combustion of coal has been well established for many years. The basic steps involved are the combustion of the coal in a furnace, the capture of the combustion heat by a boiler that produces high-temperature steam under pressure, the expansion of the steam through a turbine that drives a generator, and the condensation of the steam exhausted by the turbine (Le77). The overall thermal efficiency of this process has increased to an average of 35 percent in recent years .

Additional components required for electric power stations include heat-dissipation devices, such as evaporative cooling towers and cooling ponds; stack gas cleanup equipment, such as electrostatic precipitators and scrubbers; and coal preparation equipment (Le77).

The mineral matter in the coal forms an inorganic solid--termed ash. A portion of the mineral matter forms bottom ash or slag. The remaining portion forms fly ash which enters the flue gas stream. The radioactive materials are analogous to a solute dispersed in a solvent--the flue gas. Particulate emission control devices (including SO_x scrubbers) strip (i.e., clean) the particulates from the flue gas prior to atmospheric release. The quantity of particulates released is set by the applicable environmental standard or New Source Performance Standards. Particulate release rates (for new sources) may be less than 1.0 weight percent of the particulates in the flue gas stream.

Waste Piles

Solid waste piles (fly ash, bottom ash, slag, scrubber sludges, etc.) at coal-fired plants range in area from 80 to 100 hectares for a single 550 MW unit. In 1977 about 50 M metric tons of ash were generated by coal-fired electric generating plants in the United States. Some of the ash is stored near or on the station site; some is returned to a coal mine for disposal; and some can be used (about 10 to 20 percent of the ash.)

Control Technology for Airborne Emissions

Control technology includes cleaning the coal before it is used, process controls, and emission controls. Control technology for new plants is summarized in table 4.4-2. The control methodology is based on existing technology, current operating practices, and DOE's fuel developmental programs (EPA72, EPA77, Bo78, Si77, B&W72).

New power plants can use furnace design features minimizing fly ash formation. The options for existing stations include those for new units as well as: (1) early retirement; (2) capacity factor limitations; (3) use of improved quality coal and (4) backfitting with particulate control.

4.4.3 Emissions of Radionuclides

Calculated and measured emission rates are discussed below. The calculated emission rates are those developed for model facilities. The measured values were obtained by radioassays of samples collected by EPA's sampling program.

Calculated Emissions

Models have been developed for existing coal-fired power plants (units) and stations (RCM-10) and power plants and stations in the design stage (RCM-1) (table 4.4-3). The RCM-10 and RCM-1

Table 4.4-2 Airborne emissions control strategies for new coal-fired power plants

Fuel Block

Use high heating value fuels[a].

Avoid selection of coal containing high concentrations of uranium and thorium.

Use low mineral matter and moisture content bituminous.

Clean coal thoroughly (for example, to levels D, E, or F). (Refer to EPRI78)

Power Generation Block

Minimize unit net heat rate.

Avoid specifying overfeed stokers or dry ash, pulverized coal units.

Cyclone firing is the preferred mode of firing[b].

Slag bottom, pulverized units may be considered.

Air Block

Particulate emission control efficiency should be 99 percent or better.

If a significant degree of scrubber reheat is specified, and inline reheaters are not provided, then emission analysis is indicated to determine if a particulate control device is required in series with and following the scrubber.

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[a] Coal oil mixtures (COM) are classified as high heating value fuels.

[b] The use of a cyclone furnace may be contraindicated by the requirement to limit NO_x emissions.

models are based on a survey of data for operating units and units under construction. Eight variables relating to fuel properties, furnace design, and emission control methodology determine radionuclide release rates.

Radionuclide concentrations in the released particulates may be enriched relative to those in the mineral content of the fuel as a result of the combustion and emission control processes. Some typical enrichment factors for fossil fuel power plants are shown in table 4.4-4. The emission rates calculated for new and existing model stations incorporate enrichment factors suggested by Beck* (Be78).

The radon emission rate was determined by assuming all radon in the coal is released from the stack. Based on a literature search (Be78, Ka76), four reference coals were developed (table 4.4-5).

The RCM-10 model applies to about two-thirds of the existing units (about 800 in number--250 stations). Seventy-fire percent of these stations are believed to be located at urban and suburban sites. The model is based on a survey of Federal Power Commission data for generating stations operating as of January 1, 1970 (FPC73). The base includes data for over 300 units; Eastern bituminous is the model coal.

The RCM-1 model is representative of 300 stations built, commissioned, or planned sinced 1970. The model was developed by identifying factors controlling emission rates (table 4.4-6) and then surveying the literature for new designs (over 700) commissioned since 1959 with emphasis given to the more recent designs (306) which include environmental controls (Po75-78, NUS78, EW77). Western bituminous coal is used in this model.

Annual emission rates per operating year for model coal-fired stations (new and existing) based on the RCM-10 and RCM-1 models are listed in tables 4.4-7 and 4.4-8. The radionuclide release rate for the existing station model exceeds that for the new station model by almost a factor of three, although the existing station capacity is less than that for the new station. The difference in release rates is due to particulate control efficiency and plant net heat rate.

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*For a new plant (RCM-1), the radionuclide enrichment factor is 1.0, except for uranium, radium, lead, and polonium, which have enrichment factors of 2.0, 1.5, 5.0, and 5.0, respectively. For an existing plant (RCM-10), the radionuclide enrichment factor is 1.0 except for lead and polonium, which each have an enrichment factor of 2.0.

Measured Emissions

Samples of coal, supplementary fuels, scrubber feedstocks, slag, bottom ash, retained fly ash, scrubber sludge, and stack emissions were collected by EPA at about one percent of the nation's 1200 utility boilers located at 13 sites in 9 States. Data defining operating conditions during the stack sampling period were also obtained at each station. Factors controlling emission rates (table 4.4-9) were taken into consideration in selecting the stations to be sampled.

The initial results from this EPA sampling program are tabulated in tables 4.4-10, 11, and 12. A comparison of calculated and measured uranium-238 emission rates for several existing units is shown in table 4.4-13. The technique used to calculate the emission rates for these units was the same procedure used in the model facilities.

Analyses of fugitive emission data for coal storage and ash piles indicate the radon "exhalation rate" is less than that for soil, as reported by Beck (Be78, p.24).

4.4.4 Model Facilities

Models for existing coal-fired units and stations and new coal-fired units and stations were discussed in Section 4.4.3 and are described in table 4.4-3. The model stations consist of three RCM-10 units for existing stations and a pair of RCM-1 units for new stations.

Emissions

The radioactive emissions from the model of an existing coal fired station are shown in table 4.4-7; for the new station, in table 4.4-8.

Table 4.4-3. Model coal-fired power stations

Parameter	Units	Model New (RCM-1)	 Existing (RCM-10)
Generating Unit Model			
Type of unit	NA	SE	SE
Unit capacity	MW	550	135
Capacity factor[a]	%	65	54
Net heat rate	MJ/kWH	10.3	13.35
Furnace design	NA	(b)	(b)
Partition coefficient[c]	wt%/wt%	20/80	20/80
Particulate control	NA	LIS	ESP
Particulate control efficiency	wt%	99.0	80.0
Stack gas flow rate	m^3/s	440	ND
Stack height	m	185	97
Stack lip diameter	m	4.4	ND
Stack gas average temperature	°K	355	ND
Scrubber reheater duty	TJ/day	3.3	NA
Plume rise	m	50	50
Rank of coal	NA	Bituminous	Bituminous
Coal source region	NA	Western US	Eastern US
Heating value	GJ/ton[d]	21.46	26.5
Mineral matter content[e]	wt%	12.0	11.0
Sulfur content	wt%	1.0	3.0
Uranium concentration	ppm	1.9	1.9
Thorium concentration	ppm	5.0	5.0
Annual generation[f]	TWH/y	3.13	1.92
Annual energy requirement	PJ/y	32.2	25.6
Annual coal consumption	M ton/y	1.50	0.968
Annual total ash formation[g]	k ton/y	164	106
Annual uranium input	ton/y	2.85	1.84
Annual thorium input	ton/y	7.5	4.84

See footnotes at end of table.

Table 4.4-3. Model coal fired power stations--continued

Parameter	Units	Model New (RCM-1)	Existing (RCM-10)
Generating Station Model			
Type of generation	NA	Base load	Base load
Number of units	NA	2	3
Station capacity	MW	1100	405

NA	Not applicable.	LIS	Limestone scrubber.
SE	Steam electric.	ESP	Electrostatic precipitator.
ND	Not determined.		

[a] Capacity factor is the ratio of actual or planned generation to maximum theoretical generation. The RCM-1 value is an average value over a 30-year plant life. The RCM-10 value is a single year value.

[b] Pulverized coal, dry ash bottom.

[c] The partition coefficient is a function of furnace design. This coefficient represents the quantity of ash going into bottom ash (or slag) and the flue gas.

[d] The unit ton, wherever it appears in this table, means metric ton.

[e] Inorganic material in coal, often referred to as ash content.

[f] The annual value for RCM-1 is based on calculating the 30-year plant life value and dividing this value by 30.

[g] Ash weight is based on the assumption that the weight of the mineral matter in coal equals the weight of ash formed.

Table 4.4-4. Typical partition coefficients and enrichment factors for fossil fuel power plants

Fuel	Furnace design	Partition coefficient (wt%)		Enrichment factor for released particulates[a]			
		Bottom ash or slag	Fly ash	Uranium	Radium	Lead	Polonium
Coal	Stoker	75	25	1.0	1.0	1.0	1.0
	Cyclone	80	20	2.0	1.0	1.0	1.0
	Pulverized coal, slag bottom	50	50	2.0	1.5	5.0	5.0
	Pulverized coal, dry ash	20	80	2.0	1.5	5.0	5.0
Fuel oil		0	100	2.0	1.5	5.0	5.0
Natural gas		0	100	---------- Not applicable ----------			

[a] Actual enrichment factors are also dependent on particulate size.

Table 4.4-5. Characteristics[a] of the reference coals by rank

Coal rank	Heating value (GJ/ton)	Mineral content (wt%)	Radionuclide concentration (ppm)			
			Coal		Ash	
			Thorium-232	Uranium-238	Thorium-232	Uranium-238
Anthracite	30.16	16.7	5.4	1.5	32	9
Eastern Bituminous	26.50	11.0	5.0	1.9	99	38
Western Bituminous	21.46	12.0	5.0	1.9	42	16
Lignite	19.23	11.0	6.3	2.3	57	22

GJ/ton giga joules per metric ton.
wt% weight percent.

[a] The properties of coal are highly variable. The above values are possible ones selected from a range of values defining model coals.

Table 4.4-6. Factors which affect emissions of radionuclides from coal fired power plants

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Fuels

Types: Coal, crude oil and fuel oils.
Ranks of coal:
- Anthracite
- Bituminous
- Subbituminous
- Lignite

Coal source regions:
- Eastern United States
- Central United States
- Western United States

Furnace Designs

- Spreader stoker
- Cyclone
- Slag-bottom, pulverized coal
- Dry ash, pulverized coal

Air Handling Block

- Electrostatic Precipitators (ESPs)
- Baghouse
- Wet venturi scrubber
- Wet limestone scrubber
- Wet lime scrubber
- Mechanical precipitators

Table 4.4-7. Atmospheric emissions of radionuclides from the model existing coal-fired station (3 RCM-10 units)

Radionuclide	Emissions (Ci/y)
Uranium Series	
Uranium-238	9.7E-2
Thorium-234	9.7E-2
Uranium-234	9.7E-2
Thorium-230	9.7E-2
Radium-226	9.7E-2
Radon-222	6.7E-1
Polonium-218	9.7E-2
Lead-214	9.7E-2
Bismuth-214	9.7E-2
Polonium-214	9.7E-2
Lead-210	1.9E-1
Bismuth-210	1.9E-1
Polonium-210	1.9E-1
Actinium Series	
Uranium-235	4.6E-3
Thorium-231	4.6E-3
Protactinium-231	4.6E-3
Actinium-227	4.6E-3
Thorium-227	4.6E-3
Radium-223	4.6E-3
Radon-219	2.8E-2
Polonium-215	4.6E-3
Lead-211	4.6E-3
Bismuth-211	4.6E-3
Thallium-207	4.6E-3
Thorium Series	
Thorium-232	8.4E-2
Radium-228	8.4E-2
Actinium-228	8.4E-2
Thorium-228	8.4E-2

Table 4.4-7. Atmospheric emissions of radionuclides from the model existing coal-fired station (3 RCM-10 units)--continued

Radionuclide	Emissions (Ci/y)
Radium-224	8.4E-2
Radon-220	5.3E-1
Polonium-216	8.4E-2
Lead-212	8.4E-2
Bismuth-212	8.4E-2
Polonium-212	8.4E-2
Thallium-208	8.4E-2

Table 4.4-8. Atmospheric emissions of radionuclides[a] from the model of a new coal-fired station (2 RCM-1 units)

Radionuclide	Emissions (Ci/y)
Uranium Series	
Uranium-238	3.0E-2
Thorium-234	1.5E-2
Uranium-234	3.0E-2
Thorium-230	1.5E-2
Radium-226	2.3E-2
Radon-222	1.9
Polonium-218	1.5E-2
Lead-214	3.0E-2
Bismuth-214	1.5E-2
Polonium-214	1.5E-2
Lead-210	7.6E-2
Bismuth-210	1.5E-2
Polonium-210	7.6E-2
Actinium Series	
Uranium-235	1.5E-3
Thorium-231	7.3E-4
Protactinium-231	7.3E-4
Actinium-227	7.3E-4
Thorium-227	7.3E-4
Radium-223	9.9E-4
Radon-219	8.8E-2
Polonium-215	7.3E-4
Lead-211	7.3E-4
Bismuth-211	7.3E-4
Thallium-207	7.3E-4
Thorium Series	
Thorium-232	1.3E-2
Radium-228	2.0E-2
Actinium-228	1.3E-2
Thorium-228	1.3E-2

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[a] See footnote at end of table.

Table 4.4-8. Atmospheric emissions of radionuclides[a] from the model of a new coal-fired station (2 RCM-1 units)--continued

Radionuclide	Emissions (Ci/y)
Radium-224	2.0E-2
Radon-220	1.6
Polonium-216	1.3E-2
Lead-212	1.3E-2
Bismuth-212	1.3E-2
Polonium-212	1.3E-2
Thallium-208	1.3E-2

[a] These emission rates do not include contributions from any radionuclides in scrubber reactants.

Table 4.4-9. Parameters controlling emissions from coal-fired electric power stations

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Fuel Block

- Type of fossil fuel (coal, oil, gas, SRC, COM)
- Rank and grade of coal
- Energy content (heating value) of fuel
- Mineral matter content (ash content) of fuel
- Moisture content of coal[a]
- Sulfur content[b]
- Uranium and Thorium concentration

Energy Production Block

- Plant net heat rate[a]
- Furnace design[c]

Air Handling Block

- Particulate removal efficiency
- Scrubber reactants (if any)

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[a] All fossil fuels are burned as vapors. Before coal is combusted, the moisture content must be driven off. This step requires energy, thereby reducing boiler efficiency and in turn lowering overall efficiency.

[b] Sulfur content relates to the degree of sulfur and SO_x removal required. The quantity of lime or limestone needed by scrubbers is directly proportional to the sulfur content of the coal as fired.

[c] Stoker, cyclone, or pulverized coal unit designs.

SRC Solvent refined coal.
COM Coal oil mixture.

Table 4.4-10. Radionuclide concentrations (pCi/g) measured in coal at selected coal-fired steam electric generating stations

Radionuclide	Sampling location[a] M-1A	M-1B	M-1C	M-1D	M-1E	M-2	M-3A	M-3B	M-4A	M-4B	M-4C	M-5A	M-5B
Uranium-238	4.18	1.42	2.32	0.011	0.764	0.686	0.601	0.60	0.983	0.780	0.869	0.660	1.16
Uranium-234	4.12	1.43	2.42	0.012	0.750	0.739	0.585	0.59	0.982	0.792	0.816	0.602	1.13
Thorium-230	3.83	1.35	2.20	0.018	0.637	0.715	0.585	0.65	0.866	0.746	0.676	0.608	1.12
Radium-226	3.70	1.70	1.40	0.030	0.700	0.470	0.400	0.57	0.850	*0.59	0.92	0.620	1.00
Lead-214	3.19	1.73	2.44	0.040	0.940	0.740	0.530	0.68	0.990	0.77	0.700	0.570	0.88
Bismuth-214	2.85	1.60	2.28	0.060	0.800	0.770	0.540	0.72	0.700	0.73	0.63	0.570	0.78
Lead-210	3.11	1.90	1.74	1.33	0.520	0.490	NR	0.67	NR	1.9	NR	NR	1.00
Polonium-210	2.70	1.45	1.49	0.205	*1.12	0.875	NR	0.57	NR	0.77	NR	NR	0.68
Uranium-235	0.126	0.074	0.111	0.001	0.32	0.036	0.585	0.034	0.050	0.036	0.045	0.0962	0.072
Thorium-227	0.172	0.054	0.111	0.000	0.98	0.039	0.800	0.054	0.106	0.082	0.069	0.058	0.060
Thorium-232	0.343	0.278	0.312	0.004	0.291	0.291	0.385	0.49	0.385	0.493	0.392	0.402	0.198
Thorium-228	0.418	0.341	0.321	0.018	0.331	0.331	0.446	0.52	0.553	0.532	0.431	0.398	0.170

NR Not reported.
*Average value of two or more reported values

[a]Key:

M-1 West North Central Station.
- A Coal storage pile. Local origin.
- B Coal hopper, top. Local origin.
- C Coal hopper, bottom. Local origin.
- D Coke. Used as supplementary fuel.
- E Coal storage pile. Wyoming coal for second unit at same station.

M-2 East North Central Station.
- A Coal storage pile. Illinois origin.

M-3 South Atlantic Station.
- A Coal storage pile.
- B Pulverized coal.

M-4 Mountain Station.
- A Coal as received. Colorado origin.
- B Coal after crushing. Colorado origin.
- C Coal as fired. Colorado origin.

M-5 East South Central Station.
- A Coal storage pile. Kentucky origin.
- B Coal as fired. Kentucky origin.

Table 4.4-11. Radionuclide concentrations (pCi/g) measured in solid waste at selected coal-fired steam electric generating stations

Radionuclide	Sampling location[a] M-1A	M-1B	M-2A	M-2B	M-3A	M-3B	M-4A	M-4B	M-5A	M-5B
Uranium-238	6.85	7.01	2.99	6.36	4.96	1.43	5.62	3.93	7.66	6.76
Uranium-234	6.76	6.97	3.80	6.32	4.96	1.40	5.02	4.14	7.42	6.55
Thorium-230	7.14	8.57	3.91	5.21	4.86	1.48	4.33	4.43	6.58	5.93
Radium-226	6.6	6.9	3.3	NR	NR	NR	2.9	4.10	5.5	5.9
Lead-214	6.80	7.5	3.2	4.24	4.52	1.66	5.0	5.07	6.39	6.14
Bismuth-214	6.66	7.3	3.1	4.17	3.99	1.63	4.7	4.86	6.65	6.03
Lead-210	NR	0.57	0.30	NR	NR	NR	NR	NR	0.66	3.1
Polonium-210	5.3	[b]0.075	0.37	NR	NR	NR	NR	NR	2.00	NR
Uranium-235	0.332	0.439	0.132	0.341	0.299	0.078	0.397	0.185	0.335	0.34
Thorium-227	0.724	0.433	0.175	0.360	0.634	0.094	0.420	0.490	0.367	0.36
Thorium-232	1.12	1.36	1.28	2.04	3.56	1.46	2.49	3.05	1.2	1.04
Thorium-228	1.27	[c]1.2	1.47	2.02	3.87	1.54	2.71	3.15	1.05	0.969

NR Not reported.

[a] Key:

M-1 West North Central Station.
Cyclone fired. Local coal, 22 wt% mineral content.
A Bottom ash.
B Bottom ash.

M-2 East North Central Station.
Illinois coal.
A Bottom ash.
B Fly ash.

M-3 South Atlantic Station.
A Fly ash.
B Bottom ash.

M-4 Mountain Station.
Colorado coal.
A Fly ash.
A Bottom ash.

M-5 East South Central Station
Kentucky coal.
A Bottom ash.
B Fly ash.

[b] Two values reported, one of which was less than background: stated as a negative activity.

[c] Average of two reported values.

Table 4.4-12. Radionuclide emission rates (pCi/s) from stacks at selected coal-fired steam electric generating stations

Radionuclide	Stations[a]				
	M-1	M-2	M-3	M-4	
				<1.0μ	>1.0μ
Uranium-238	1180	278	36.9	1.64	3.20
Uranium-234	1175	350	39.3	1.66	3.17
Thorium-230	73.8	199	14.3	1.05	2.96
Radium-226	257	202	10.4	0.73	0.34
Lead-210	1380	734	65.9	6.5	2.42
Polonium-210	3326	698	54.6	6.3	1.38
Uranium-235	60	19.2	1.59	0.009	0.20
Thorium-227	NR	20.6	1.2	0.16	0.05
Thorium-232	39.3	73.2	9.9	0.56	1.77
Thorium-238	35.2	84.7	14.8	0.79	1.69

[a] M-1 West North Central Station (510 MW).
M-2 East North Central Station (450 MW).
M-3 South Atlantic Station (125 MW).
M-4 Mountain Station (12.5 MW).

NR Not reported.

Table 4.4-13. Comparison of measured and calculated uranium-238 emission rates from selected coal fired stations

Station[a]	Uranium-238 activity in coal (fCi/g)[b]	Power level[c] (MW)	Coal feed rate (kg/s)	Furnace design	Particulate control	Uranium-238 emission rate			Percent released[f] (%)	Percent difference[g] (%)
						Measured[d] (pCi/s)	Calculated[e]			
							No enrichment (pCi/s)	With enrichment (pCi/s)		
M-1	[h]2640	510	68.5	Cyclone	LIS	1180	560	1120	0.65	5.4
M-2	686	450	58.4	Cyclone	ESP	278	80	160	0.69	73.0
M-3	600	125	10.3	(i)	ESP + MP	36.9	30.4	60.8	0.60	65.0
M-4	983	12.5	1.6	Stoker	Baghouse	4.8	3.9	(j)	0.31	23.0

[a] See table 4.4-12 for station key.

[b] A fCi (femtocurie) equals 1E-15 curies.

[c] Power level during test.

[d] Two sigma error not tabulated.

[e] All calculated release rates are based on a control efficiency of 99%.

[f] Ratio of emission rate to activity feed rate. The emission rate used is the measured emission rate and is not a measure of particulate control efficiency.

[g] Percent difference is the quotient of the absolute difference between measured and enriched values divided by the smaller value.

[h] Average of 4180, 1420, and 2320 fCi/g.

[i] Pulverized coal, slag bottom.

[j] No enrichment factor applied to a stoker.

LIS Wet limestone SO_x scrubber.
ESP Electrostatic precipitator.
MP Mechanical precipitator.

4.4.5 Health Impact Assessment of Model Facilities

Estimates of working level exposures and annual radiation doses resulting from the radioactive emissions from the model facilities are shown in tables 4.4-14, 4.4-15, and 4.4-16. Because these coal fired stations are located on sites exhibiting a wide range of characteristics, estimates are presented for the model station located at an urban, suburban, rural, and remote site and the population distribution within 80 kilometers of these sites is shown in table 4.4-17.

Tables 4.4-18 and 4.4-19 present estimates of the individual lifetime risks and numbers of fatal cancers to the population resulting from particulate doses at each of the generic sites for each model station. The urban site is a conservative selection and estimates for this site represent an upper limit of the potential health impacts to a regional population.

The risks from radon-222 are not significant in comparison to the risks from radionuclides in particulate materials.

For the new coal fired station, the lifetime risk of a fatal cancer to the ost exposed group of individuals ranged fom about 1 x 10^{-5} to about 6 x 10^{-5}. The number of fatal cancers per year of station operation in the regional population ranged from 8 x 10^{-5} to 0.2.

For the existing coal fired station, the lifetime risk to the most exposed group of individuals ranged from about 6 x 10^{-5} to 7 x 10^{-4}. The number of fatal cancers per year of station operation in the regional population ranged from 4 x 10^{-4} to 1.5.

Table 4.4-14. Working level exposures from radon-222 emissions from the model coal fired stations

Site	New Stations Maximum Individual (WL)	New Stations Regional Population (person-WL)	Existing Stations Maximum Individual (WL)	Existing Stations Regional Population (person-WL)
Urban	5.4E-9	2.4E-2	4.0E-9	1.3E-2
Suburban	6.8E-9	1.9E-3	4.4E-9	7.9E-4
Rural	9.7E-9	5.0E-4	6.3E-9	2.2E-4
Remote	6.7E-9	4.8E-6	5.2E-9	2.1E-6

Table 4.4-15. Annual radiation doses from radioactive particulate emissions from the model "existing" coal-fired station[a]

Organ	Urban site Maximum Individual (mrem/y)	Urban site Population (person-rem/y)	Suburban site Maximum Individual (mrem/y)	Suburban site Population (person-rem/y)
Lung	6.7	19000	9.2	1050
Bone	8.3	11000	11.1	940
Kidney	4.7	5200	5.9	460
Liver	4.7	4600	5.8	310
Thyroid	5.5	5500	6.8	360
G.I. tract	4.2	3700	5.1	220
Other soft tissue	5.5	5700	6.9	370

Organ	Rural site Maximum Individual (mrem/y)	Rural site Population (person-rem/y)	Remote site Maximum Individual (mrem/y)	Remote site Population (person-rem/y)
Lung	15.0	300	8.1	2.7
Bone	62.0	290	12.0	7.1
Kidney	19.0	150	3.5	2.0
Liver	13.0	97	2.2	1.0
Thyroid	17.0	110	2.9	1.4
G.I. tract	8.5	67	1.4	0.4
Other soft tissue	17.0	120	2.9	1.4

[a] This station contains three RCM-10 model units.

Table 4.4-16. Annual radiation doses from radioactive particulate emissions from the model "new" coal-fired station[a]

Organ	Urban site		Suburban site	
	Maximum Individual (mrem/y)	Population (person-rem/y)	Maximum Individual (mrem/y)	Population (person-rem/y)
Lung	0.88	2600	1.1	190
Bone	1.6	1300	2.1	160
Kidney	0.79	650	1.0	95
Liver	0.77	580	0.9	43
Thyroid	0.9	650	1.1	49
G.I. tract	0.67	470	0.8	31
Other soft tissue	0.9	680	1.1	51

Organ	Rural site		Remote site	
	Maximum Individual (mrem/y)	Population (person-rem/y)	Maximum Individual (mrem/y)	Population (person-rem/y)
Lung	2.1	51	0.79	0.47
Bone	16.0	50	2.1	1.5
Kidney	4.6	31	0.61	.43
Liver	2.4	16	0.31	.17
Thyroid	2.7	15	0.36	.18
G.I. Tract	1.4	10	0.18	.063
Other soft tissue	2.8	16	0.37	.18

[a] This station consists of two RCM-1 units.

Table 4.4-17. Population distribution of model coal fired power station sites

Site[a]	Population[b]
Urban (Site A)	17,100,000
Suburban (Site B)	2,490,000
Rural (Site D)	592,000
Remote (Site F)	11,900

[a] Sites are described in Appendix A.
[b] Within a radius of 80 km.

Table 4.4-18. Individual lifetime risks and number of fatal cancers due to radioactive particulate emissions from the model "existing" coal fired plant

Site	Individual lifetime risks Maximum individual	Average individual	Expected fatal cancers per year of operation (Fatal cancers)
Urban	7.0E-5	6.2E-6	1.5
Suburban	9.2E-5	2.8E-6	1.0E-1
Rural	7.1E-4	3.5E-6	3.0E-3
Remote	6.3E-5	2.5E-6	4.3E-4

Table 4.4-19. Individual lifetime risks and number of fatal cancers due to radioactive particulate emissions from the model "new" coal-fired station

Site	Individual lifetime risks Maximum individual	Average individual	Expected fatal cancers per year of operation (Fatal cancers)
Urban	1.1E-5	8.6E-7	2.1E-1
Suburban	1.4E-5	4.8E-7	1.7E-2
Rural	5.6E-5	5.7E-7	4.8E-3
Remote	8.9E-6	4.7E-7	8.0E-5

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4.5 Metal and Nonmetal Mining and Milling

Uranium and thorium and their daughter products including gaseous radon radionuclides are naturally occurring constituents of the earth's crust. Therefore, any activity which involves the disturbance of the earth's surface can result in some release of these radionuclides to the atmosphere. Because mining and milling activities involve the handling and processing of large quantities of ore removed from below the earth's surface, these activities have been assessed to determine their potential for release of radioactive materials to the atmosphere and the health impact resulting from these releases.

The task of identifying industrial segments to be assessed, characterizing their emissions, and assessing their health impacts is extremely difficult because there are approximately 15,000 mines of many diverse sizes and types. Therefore, to reduce this assessment to a manageable size, only certain types of industries were selected.

The industries selected, iron, copper, zinc, clay, limestone, fluorspar, and bauxite, are relatively large industries and are considered to have the greatest potential for emitting radioactive materials into the ambient air. Some of the criteria for selecting these industries were: number of mines, production rate, working level concentrations of radioactive materials in the mines, ventilation rate, and size of waste tailings (table 4.5-1).

4.5.1 General Description

Iron

Crude iron ore production in the United States amounted to 215 million tons in 1975. Open pit mines produced 96 percent of total output and nearly all the ore was shipped to beneficiation plants. The average iron content of all crude ore mined in 1975 was 33 percent and the average iron content of all usable ore produced was 61 percent. The average annual production of crude ore was 3.2 million tons per mine.

Twenty States have iron ore deposits but the Lake Superior States of Michigan, Minnesota, and Wisconsin produced 85 percent of the ore mined in 1975.

The iron industry has a projected annual growth rate of 1.0 to 2.0 percent over the period 1975 to 2000.

Table 4.5-1. Number, type, and capacity of selected mining industries, United States, 1975

Mining Industry	Number and type of mines		Production (thousand short tons)	Radon daughter concentrations (Working Level)[a]	Radon concentrations at 50% equilibrium (pCi/L)
	Open pit	Underground			
Iron	57	11	497,000	0.14-0.90	140
Copper	46	15	959,000	0.09-0.21	30
Zinc	(b)	36	11,400	0.07-1.40	146
Clay	1,317	(c)	80,500	0.10-0.46	56
Limestone	2,900	(c)	971,000	0.05-0.16	22
Flurospar	1	14	593	0.30-2.20	250
Bauxite	11	1	16,600	0.07-1.40	146

[a] Source: Surveys by Mine Safety and Health Adminstration Inspectors (MESA78).

[b] A small, undetermined number are open pit mines.

[c] A small, undetermined number are underground mines.

Copper

Approximately 263 million short tons of crude copper ore were produced in the United States in 1975. This total yielded only 1.41 million short tons of copper because almost all of the crude ore is waste, containing only 0.75 percent copper. Usable ore obtained from beneficiation contains 25 percent copper.

Six States, Arizona, Utah, New Mexico, Montana, Nevada, and Michigan accounted for 98 percent of the total copper production in the United States in 1975. Arizona produced 58 percent of the total recoverable copper or 813 thousand short tons.

The averge annual output of the significant mines in the United States was about 10 million tons of crude ore per mine with open pit mines accounting for 80 percent of mine production.

The growth rate of the copper industry is variable--1.0 percent to 15 percent--depending on the state of the Nation's economy.

Zinc

About 500 thousand short tons of zinc were produced in the United States in 1975. The major ore producing States are Tennessee, New York, Missouri, Colorado, and Idaho, with the States east of the Mississippi River accounting for one-half of the total United States production.

Essentially all zinc mines are underground. The crude domestic ore contains approximately 4 percent zinc, 7 percent usable ore and 93 percent waste. The usable ore contains approximately 55 percent zinc. The average annual production of crude ore per mine is about 500 thousand short tons.

The zinc mining industry does not appear to have any significant growth rate in the immediate future.

Clay

Clay is produced in 47 States with Georgia, Texas, and Ohio the leading producers. In 1975, 1317 clay mines in the United States produced 49.4 million tons of clay. About 90 percent of clay production comes from surface mines.

The average annual production per mine was 40 thousand short tons. There is no waste in clay mining.

Clay is categorized as kaolin, ball clay, fire clay, bentonite, fuller's earth, or common clay and shale--the latter category is produced by about 60 percent of the mines.

An annual growth rate of 3.5 percent has been projected for this industry.

Limestone

The crushed limestone industry converts naturally occurring limestone deposits to a form that is predominantly used by the construction industry. Processing plants produced about 6.7×10^7 short tons of crushed limestone in 1975 from more than 2900 quarries or underground mines in 46 States. Illinois, Texas, Pennsylvania, Missouri, and Ohio accounted for 38 percent of the 1975 limestone production.

Crushed limestone plant production ranges from 300 thousand to 2.4 million short tons per year, with average plant production about 600 thousand short tons per year. There is no waste associated with the product.

Since 1945 the total output of crushed stone has multiplied seven times. The annual growth rate of this industry is approximately 5 percent.

Fluorspar

Fluorspar producing mines in the United States are located in Illinois, Utah, Texas, Montana, Nevada, Kentucky and New Mexico.Mine production in 1975 amounted to about 375 thousand short tons, with about 75 percent of the production coming from Hardin County, Illinois. The ore was beneficiated by 8 plants resulting in 132 thousand short tons of recovered material. Fluorspar ore is 70 percent waste. Most mines are underground and the average production per mine was 15 thousand short tons.

The estimated annual growth rate for the industry is 3 percent.

Bauxite

The United States imported 86 percent of the bauxite consumed in 1975, mining only 14 percent or 2 million short tons in 12 mines located in Arkansas, Georgia, and Alabama. Arkansas produced 87 percent of the bauxite mined in 1975.

Bauxite mines in the United States are open pit mines except for 1 underground mine in Saline County, Arkansas. The ore is 45 percent waste. The industry's growth rate is minimal.

4.5.2 Process Description

The three basic processes associated with these industries are mining, milling, and smelting. However, an assessment of the smelting process has not been included because sufficient measurement data on emissions are not available at this time to assess the effect of this process. Future assessments will include emissions from smelting or similar type operations.

Mining

Mining is the initial step in the acquisition of ore and is performed both underground and above ground. Extraction techniques vary with minerals and locations of deposits. Mining includes operations such as drilling, blasting, loading, hauling and removal of rock. Under some conditions, bulldozing and dredging may be substituted for drilling and blasting. These operations are carefully planned and executed because of high costs and the results affect the operating costs of subsequent steps. A single blast can fragment thousands of tons of ore. Large shovels with capacities of 7 to 13 cubic meters (9-17 cubic yeards) are used for loading operations. Large capacity trucks (50-350 tons) are frequently used for hauling; however, railroad trains containing as many as 160 cars are often used to haul ore from remote mines to mills (e.g., 50 miles). Each of the foregoing mining operations and transfer points has the potential for producing radioactive emissions.

Milling

Milling procedures, which are often performed at or near the mine, enhance mineral recovery. Some of the procedures are crushing, grinding, screening, blending, concentrating, classifying, briquetting, sintering, and agglomerating. The physical and chemical properties of the crude ore determine the milling method. After recovery of the desirable concentrate, a large volume of tailings remains as residue or waste from the milling procedures.

The source of particulate emissions in the processing of ore are those locations, including tailing piles, where the ore is in motion or the atmosphere surrounding the ore or its products is in motion. These locations are mining, crushing, screening, grinding, bagging, drying, drilling, loading, hauling, blasting and all points of transfer such as truck and railroad car loading points.

Control Technology

Dust is produced at many of the stages in the mining and milling processes. Methods to reduce emissions include wet dust suppression, dry collection and a combination of the two. In wet dust suppression, moisture is introduced in the material flow causing fine particulate matter to remain with the material flow rather than becoming airborne. Dry collection involves hooding and enclosing dust-producing points and exhausting emissions to a collection device. Both methods may be combined at different stages throughout the processing plant. Structures enclosing process equipment are also effective control methods.

4.5.3 Emissions of Radionuclides

Because little data are presently available on radioactive emissions from the nonuranium, nonphosphate mining and milling industry, most of the emissions data used in this section are rough estimates (first approximations) calculated primarily from the uranium content of the ore. In a few cases, however, the initial results of a field study program (described below) were used and these data are identified in the appropriate tables.

All calculations and assumptions used in developing emission estimates are presented in the appendices. The general approach used in these calculations was to estimate the uranium content of the ore and to extrapolate information on emissions from the uranium mining and milling industry to those of nonuranium mining and milling activities based on the uranium content of the ore (i.e., normalizing to uranium content). In these calculations, the assumption was made that uranium-238 was in equilibrium with its decay products and further, that the same specific activity of uranium and decay products present in the ore was also present in the dust emissions.

For all of the industries evaluated, with the exception of copper, a single best estimate of the uranium-238 content of the ore was made. For the copper industry, the uranium-238 content is expressed as a range because information in the literature indicated that some copper deposits contain elevated levels of the uranium-238 decay series (Fi76).

Since there is a lack of data on the thorium content of ores, the emissions data used in this section are limited to members of the uranium decay series. Information is being developed on the thorium content of ores and future assessments will include impacts from emissions of thorium and its decay products.

Through measurement studies EPA is presently developing data on radioactive emissions from mining and milling operations. Measurements are being made of the emissions of radon-222 and radioactive particulates from mine vents, mill and smelter stacks and vents and waste piles at several facilities for each of the industries listed in table 4.5-1. In a limited number of cases, the preliminary data from these studies are used in this report. However, for most facilities the data will not be available until late in 1979 or early 1980. These studies, which are quite limited in relation to the large size of the mining and milling industry, will be useful only in characterizing the emissions from these industries in a general way. However, these studies should identify any significant problems, existing on a broad scale within the industries studied, with emissions of radioactive materials.

4.5.4 Model Facility

To estimate the radioactive emissions and health impacts from the selected mines, mills, and tailings piles, a model mine, mill, and tailings piles for each industry were developed by assigning values for the parameters which are important in assessing impacts (table 4.5-2).

Emissions

The atmospheric emissions of radioactive material from the model mines, mills, and tailing piles are listed in table 4.5-3.

4.5.5 Health Impact Assessment of Model Mine and Mill

Tables 4.5-4 and 4.5-5 estimate annual working level exposures and radiation doses resulting from radioactive emissions from the model mines, mills and tailings piles. The estimates for the copper mining and milling industry are for a remote, sparsely populated site in the Southwest; the site has minimal impact on the general population (Site E, Appendix A). The estimates for all other industries were based on a rural site in south central United States (Site D, Appendix A). The individual lifetime risks and number of fatal cancers estimated to result from the radioactive doses for the selected industries are shown in tables 4.5-6 and 4.5-7.

Table 4.5-2. Model mines, mills, and tailing piles for selected industries

Parameter	Value for selected industry						
	Iron	Copper	Zinc	Clay	Limestone	Fluorspar	Bauxite
Mine							
Type of mine	Open pit	Open pit	Underground	Open pit	Open pit	Underground	Open pit
Ore process rate (metric tons/y)	2.72E+6	9.08E+6	4.54E+5	3.63E+4	5.45E+5	1.36E+4	1.82E+5
Operating days per year	365	365	365	365	365	365	365
Mine lifetime (years)	100	100	100	100	100	100	100
Ore grade	35% Fe_2	0.75% Cu	4% Zn	100%	100%	30%	55% Al_2O_3
Uranium in ore (pCi/g)	0.67	0.33-18	1	2	0.67	0.67	2.67
Effective release height (meters)	10	10	10	10	10	10	10
Mill							
Ore process rate (metric tons/y)	2.72E+6	9.08E+6	4.5E+5	3.63E+4	5.45E+5	1.36E+4	1.82E+5
Operating days per year	250	250	250	250	250	250	250
Mill lifetime (years)	50	50	50	50	50	50	50
Effective stack height (meters)	20	20	20	20	20	20	20
Particulate emission rate (g/s)	6.3	1	0.16	0.14	0.4	0.12	0.15
Tailings							
Waste generated (metric tons/y)	1.77E+6	8.9E+6	4.21E+5	None	None	[a]1.91E+5	[a]1.63E+6
Area of waste pile (hectares)	530	270	44	None	None	6.4	17
Average depth (meters)	12.2	12.2	12.2	NA	NA	1.5	4.88

[a] Per 20 year period.
NA Not applicable.

Table 4.5-3. Atmospheric emissions of radionuclides from the model mines, mills, and tailing piles of selected industries

Source	Emissions			
	Iron	Copper	Zinc	Clay
Mine Radon-222 (Ci/y)	67	27-1518	115	18
Mill Uranium-238 (μCi/y)[a]	91	7-396	3.6	6
Tailings pile Radon-222 (Ci/y)	160	(b)-1500	(b)	(b)

Source	Emissions		
	Limestone	Fluorspar	Bauxite
Mine Radon-222 (Ci/y)	7	0.2	10
Mill Uranium-238 (μCi/y)[a]	6	1.7	9
Tailings pile Radon-222 (Ci/y)	(b)	(b)	7

[a] The same release rate also applies to each member of the uranium series decay chain released in the dust emissions.

[b] Does not differ from background.

Table 4.5-4. Working level exposures from radon-222 emissions from model mines and tailings piles for selected industries[a]

Source	Maximum individual (WL)	Regional population (person-WL)
Mining operations:		
Iron	1.5E-5	2.4E-1
Copper (Lowest)	1.1E-5	5.2E-3
(Highest)	6.3E-4	2.9E-1
Zinc	2.6E-4	4.1E-1
Clay	3.5E-5	6.4E-2
Limestone	5.8E-6	2.5E-2
Fluorspar	4.6E-7	7.1E-4
Bauxite	8.3E-6	3.5E-2
Tailings disposal area:		
Iron	3.5E-5	5.7E-1
Copper (Lowest)	-	-
(Highest)	6.2E-4	2.9E-1
Zinc	-	-
Clay	NA	NA
Limestone	NA	NA
Fluorspar	-	-
Bauxite	6.1E-6	2.6E-2

[a] Essentially no radon is released in the milling operation.
NA Not available

Table 4.5-5. Annual radiation doses from radioactive particulate emissions from the model mills for selected industries

Organ	Iron		Copper		Zinc	
	Maximum individual (mrem/y)	Population (person-rem/y)	Maximum individual (mrem/y)	Population (person-rem/y)	Maximum individual (mrem/y)	Population (person-rem/y)
Lung	0.85	1.8E-1	0.12-6.9	4.6E-4-2.6E-2	3.4E-2	7.2E-3
Bone	1.26	1.9E-1	0.18-10	1.2E-3-6.5E-2	5.0E-2	7.6E-3
Kidney	0.43	1.2E-1	0.06-3.5	4.7E-4-2.6E-2	1.7E-2	4.8E-3
Liver	0.23	5.7E-2	0.03-1.9	1.7E-4-9.7E-3	9.1E-3	2.2E-3
Thyroid	0.40	8.1E-2	0.06-3.2	3.2E-4-1.8E-2	1.6E-2	3.2E-3
G.I. Tract	0.16	3.8E-2	0.02-1.3	9.3E-5-5.2E-2	6.4E-3	1.5E-3
Other soft tissue	0.41	8.3E-2	0.06-3.3	3.3E-4-1.8E-2	1.6E-2	3.3E-3

Organ	Clay		Limestone		Fluorspar	
	Maximum individual (mrem/y)	Population (person-rem/y)	Maximum individual (mrem/y)	Population (person-rem/y)	Maximum individual (mrem/y)	Population (person-rem/y)
Lung	5.6E-2	1.2E-2	5.6E-2	1.2E-2	9.4E-3	3.4E-3
Bone	8.3E-2	1.3E-2	8.3E-2	1.3E-2	1.4E-2	3.6E-3
Kidney	2.8E-2	8.0E-3	2.8E-2	8.0E-3	4.7E-3	2.3E-3
Liver	1.5E-2	3.7E-3	1.5E-2	3.7E-3	2.5E-3	1.1E-3
Thyroid	2.6E-2	5.3E-3	2.6E-2	5.3E-3	4.4E-3	1.5E-3
G.I. Tract	1.1E-2	2.5E-3	1.1E-2	2.5E-3	1.8E-3	7.2E-4
Other soft tissue	2.7E-2	5.5E-3	2.7E-2	5.5E-3	4.5E-3	1.6E-3

Organ	Bauxite	
	Maximum individual (mrem/y)	Population (person-rem/y)
Lung	8.4E-2	1.8E-2
Bone	1.2E-1	1.9E-2
Kidney	4.3E-2	1.2E-2
Liver	2.3E-2	5.6E-3
Thyroid	3.9E-2	8.0E-3
G.I. Tract	1.6E-2	3.8E-3
Other soft tissue	4.0E-2	8.3E-3

Table 4.5-6. Individual lifetime risks due to radioactive emissions from model mines, mills, and tailing piles for selected industries

Industry		Mines		Mills	
		Maximum individual	Average individual	Maximum individual	Average individual
Iron		2.2E-5	5.9E-7	7.1E-6	2.4E-09
Copper	(Lowest)	1.7E-5	2.2E-7	1.0E-6	1.5E-10
	(Highest)	9.4E-4	1.2E-5	5.7E-5	8.5E-09
Zinc		3.9E-4	1.0E-6	2.9E-7	9.6E-11
Clay		4.5E-5	1.6E-7	4.7E-7	1.6E-10
Limestone		8.7E-6	6.2E-8	4.7E-7	1.6E-10
Fluorspar		6.9E-7	1.8E-9	7.9E-8	4.5E-11
Bauxite		1.2E-5	8.9E-8	7.1E-7	2.4E-10

Industry		Tailings	
		Maximum Individual	Average Individual
Iron		5.2E-5	1.4E-6
Copper	(Lowest)	-	-
	(Highest)	9.3E-4	1.2E-5
Zinc		-	-
Clay		-	-
Limestone		-	-
Fluorspar		-	-
Bauxite		9.2E-6	6.6E-8

Table 4.5-7. Estimated number of fatal cancers from model mines, mills, and tailings piles

Industry	Estimated fatal cancers per year of operation		
	Mines (Fatal cancers)	Mills (Fatal cancers)	Tailings (Fatal cancers)
Iron	5.0E-3	2.0E-5	1.2E-2
Copper (Lowest)	1.1E-4	8.0E-8	-
(Highest)	6.2E-3	4.5E-6	6.1E-3
Zinc	8.6E-3	7.9E-7	-
Clay	1.3E-3	1.3E-6	-
Limestone	5.2E-4	1.3E-6	-
Fluorspar	1.5E-5	3.7E-7	-
Bauxite	7.4E-4	2.0E-6	5.5E-4

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4.6 Radon from Water

4.6A Geothermal Power Plant Sites

4.6A.1 General Description

Geothermal energy, the energy contained in the heat of the earth, is a resource receiving increasing scientific and public attention. Geothermal energy in the form of hot springs has been used for years to provide "healthful and refreshing warm baths" at spas and resorts. More recently, this energy source is being investigated and developed for the production of electric power.

At present, power applications of geothermal resources are confined to one locality, The Geysers, a geothermal site in California 80 miles north of San Francisco. There are currently 12 power plants at The Geysers with an electrical capacity of about 600 MW(e). The geothermal resources at other areas are now being evaluated by the Department of Energy and may be exploited for power production in the near future.

4.6A.2 Process Description

There are three primary types of geothermal resources: 1) hydrothermal convection systems, 2) hot igneous systems, and 3) conduction-dominated systems.

Hydrothermal Convection Systems

Hydrothermal convection systems are created when a source of heat, usually molten rock or magma, comes in contact with a permeable rock formation containing water which expands and rises upward as it is heated by the molten rock below. Above the permeable rock layer an impermeable layer of rock traps the superheated water. If the impermeable rock layer contains cracks through which water can rise, the fluid will appear on the earth's surface either as steam in a vapor-dominated system or as hot water in a liquid-dominated system.

At present, only one vapor-dominated system in the United States produces power commercially--the system at The Geysers in California. At this location, approximately 10 steam-producing wells have been drilled into the steam-bearing layer of rock and tied together to produce enough steam to drive a turbine. Noncondensable gases, such as radon and hydrogen sulfide, are discharged to the atmosphere and condensables are either discharged to the surface water system, or more commonly, are pumped down wells to recharge the geothermal reservoir. When the turbine is shut down, the wells supplying that turbine are usually vented directly to the atmosphere to prevent condensate from building up in the well. A well lasts about 15 years before it is depleted.

Only two other vapor-dominated systems have been identified in the United States. The first is the Mud Volcano System in Yellowstone National Park, Wyoming, and the other is an unconfirmed system in Mt. Lassen National Park, California. Neither of these systems is used for commercial purposes.

About 63 high- and medium-temperature (90° C) hot water or liquid-dominated systems have been identified in the United States. Although none of these systems is currently being used for the commercial production of power, several experimental plants are under development.

Hot Igneous Systems

Hot igneous systems consist of magma (molten rock occurring near the surface of the earth) and hot dry rock (the solidified margins around the deposits of magma and the overlying roof rock). These systems differ from hydrothermal convection systems in that the rock formations are generally not permeable enough to trap water. The recovery of geothermal energy directly from magma or from hot dry rock is not yet feasible, although some development work is under way.

Conduction-dominated Systems

Most of the earth's heat is transferred to the surface through the process of conduction through solid rock. The first of these processes, called "normal gradient," is not presently being used as a source of geothermal energy.

The second process is called geopressured geothermal reservoirs. Like the normal gradient, the temperature of a geopressured resource increases with depth at a constant, normal rate. The geopressured reservoir differs in being a formation of methane-saturated water trapped in layers of sand and shale beneath impermeable rock. The weight of the sediment creates extremely high water temperatures and pressures. A geopressured zone is known to exist beneath an area extending from the Rio Grande in Texas to the mouth of the Pearl River in Louisiana and from several kilometers inland to the edge of the continental shelf. This resource is presently in the experimental stage.

4.6A.3 Emissions of Radionuclides

The primary source of radioactivity from geothermal installations is due to the natural uranium decay chain, most commonly radium-226 and radon-222. Radium-226 would tend to remain a water pollution or land disposal problem because it is nonvolatile; therefore, the primary air pollutant is radon-222, a gas.

Radon-222 is released when the water or steam from a geothermal resource contacts the air. In an electric power-generating system, this could occur at the wellhead before the well is connected to a turbine-generator or where the effluent (steam and/or water) from a power-generating system is discharged to a surface stream or a cooling tower. In a closed-cycle system, where all the effluent from power generation is returned to the geothermal resource through a deep well, radon-222 is not a problem unless noncondensable gases are vented to the atmosphere.

At the present time, radon-222 measurement data are available for only a few geothermal sites such as The Geysers, the Niland Geothermal Test Facility, the Bureau of Reclamation's East Mesa Facility, the Chevron Oil Company Project in Heber, California, and one geopressure well at Vermillion Bay, Louisiana. These data which are primarily in the form of concentrations in air ejector gases, noncondensable gases, and steam condensate, are summarized in table 4.6A-1.

The variation in these data highlights the expected change in radon-222 over time, its dependency on flow conditions, and in some part, on the sample collection method. Relating radon-222 to noncondensable gases, steam condensate, or unflashed brine is dependent upon the objectives of the technical study which produced these data.

At the present time, radon-222 emission estimates are available only for The Geysers site since other sites are still in the development stage.

4.6A.4 Typical Facility

In order to estimate health impacts from geothermal power sites, The Geysers site was adopted as a typical site and the parameters from this site were used in assessing impacts (table 4.6A-2). Atmospheric emissions from the typical facility are shown in table 4.6A-3.

Table 4.6A-1. Summary of radon-222 emissions data from geothermal power sites

Site	Radon-222 data	References
The Geysers	- 1.4 Ci/day released from 11 power plants (1975)	(AN78)
	- 2.1 pCi/L noncondensable gases at wellhead	(LFE75)
	- 2400-6800 pCi/L noncondensable gases at air ejector	(LFE75)
	- 16500-18000 pCi/L steam condensate	(St75)
Niland	- 284-526 pCi/L noncondensable gases	(LFE75)
	- 7.1-11.0 pCi/L of steam condensate	(ORP76)
	- 2000-6100 pCi/L steam condensate	(St75)
	- 830-1150 pCi/L noncondensable gases, Woolsey well	(Ro78)
	- 535-644 pCi/L noncondensable gases, Magmafax #1	(Ro78)
	- 440-660 pCi/L brine Magmafax #1	(St75)
East Mesa Bureau of Reclamation	- 1240 pCi/kg brine, Well 5-1	(ORP76)
	- 1287 pCi/kg brine, Well 6-1	(ORP76)
	- 69 pCi/L brine	(St75)
East Mesa Republic Geothermal	- 1095-1262 pCi/L noncondensable gases	(Ro78)
Heber	- 3200-4300 pCi/L noncondensable	(Ro78)
Vermillion Bay, Louisiana	- 10 - 37 pCi/L noncondensable gases	(Ro78)

Table 4.6A-2. Typical geothermal power site

Parameter	Value
Electrical capacity	600 Mw(e)
Availability factor	0.9
Number of units	12
Average electrical capacity per unit	50 MW(e) (Range 11-106)
Stack height	10 meters

Table 4.6A-3 Atmospheric emissions of radionuclides from typical geothermal power sites (AN75)

Radionuclide	Emissions[a] (Ci/y)
Radon-222	540

[a] 45 Ci/y from each unit.

Since there is presently only one geothermal power site in operation (Geysers site), it is difficult to know how representative the impacts from this site will be of the impacts from future sites. It is important therefore that any interpretation of the impact assessment for this category take this limitation into consideration.

4.6A.5 Health Impact Assessment of a Typical Geothermal Power Site

The estimated working level exposures that would result from radon-222 emissions from a typical geothermal power site are listed in table 6.4A-4. These estimates are for a site similar to that of The Geysers (80 km population, 310,000). The typical geothermal power site consists of 12 electrical generating units, each releasing an equal amount of radon-222. These units are widely dispersed over a large site; two units are usually located in close proximity to each other. The working level exposures to the highest exposed group of individuals were calculated at a location 500 meters from the two units in the predominant wind direction assuming a combined release of 90 Ci/y for the two units.

Table 6.4A-5 estimates the individual lifetime risks and number of fatal cancers resulting from these exposures. The lifetime risk of lung cancer to the highest exposed group of individuals is about 6.0E-4 and the lifetime risk of lung cancer to the average individual in the region is estimated to be about 7.0E-6. The annual number of fatal lung cancers is estimated to be about 0.03.

Table 6.4A-4. Working level exposures from radon-222 emissions from a typical geothermal power site

	Maximum individual (WL)	Regional population (Person-WL)
Geothermal power site	4.3E-4	1.5

Table 6.4A-5. Individual lifetime risks and number of fatal cancers due to radon-222 emissions from a typical geothermal power site

	Individual lifetime risks		Expected fatal cancers per year of operation (Fatal cancers)
	Maximum individual	Average individual	
Geothermal power site	6.5E-4	7.3E-6	3.1E-2

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4.6B Ground Water Treatment Plants

4.6B.1 General Description

The rocks, sands and clays that make up aquifers all contain concentrations of naturally-occurring radium-226. When this radioactive isotope decays, it forms radon-222 as its daughter. Radon-222 is a radioactive gas with a half-life of 3.8 days. As a gas, radon is able to migrate freely into water in the aquifer. The concentration of radon-222 in ground water normally far exceeds the radium-226 concentration. When the aquifer is tapped to withdraw the water, the radon gas will be withdrawn also.

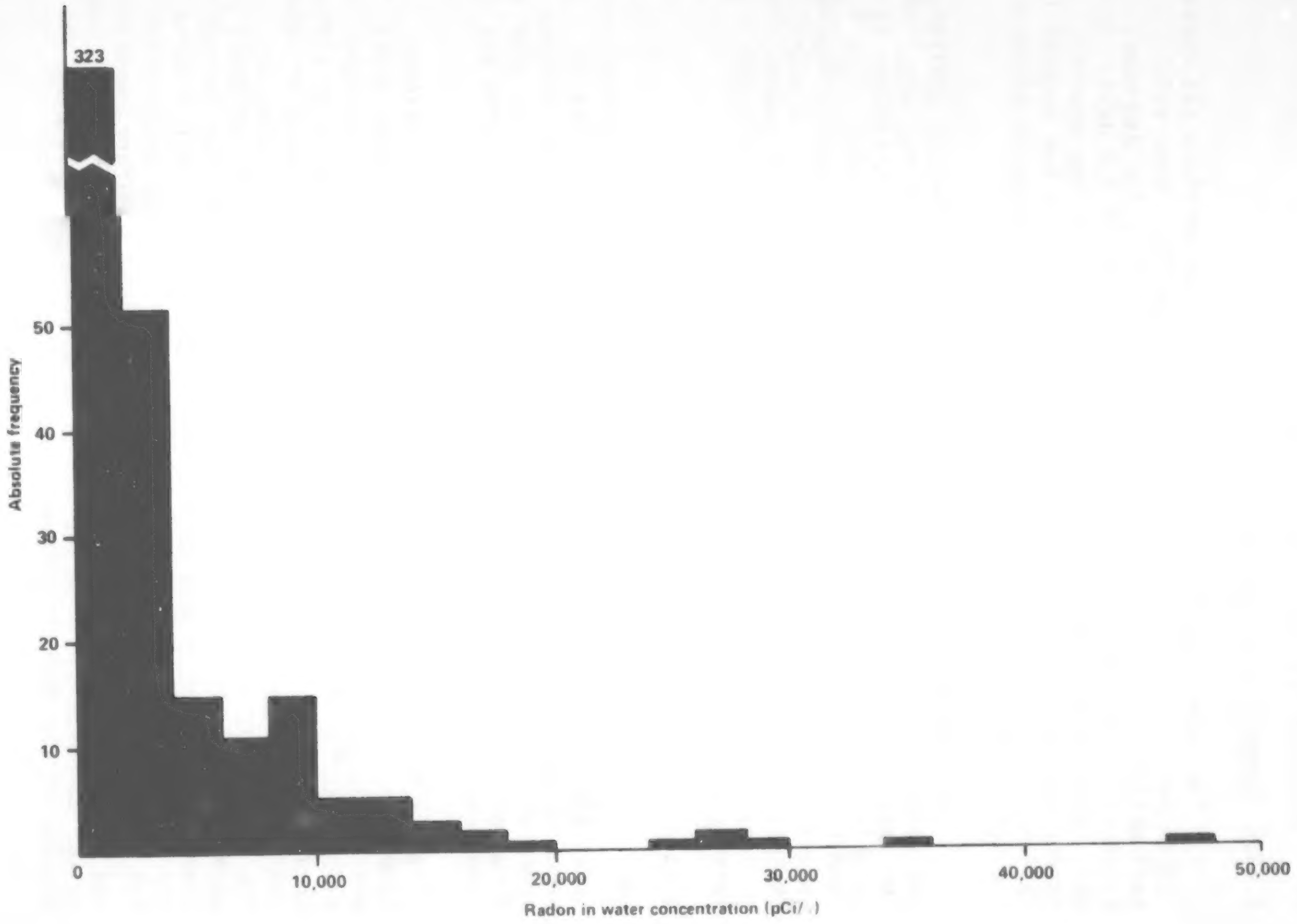
When water containing radon is withdrawn from the aquifer, the radon is available for release to the air. This may occur at the water treatment plant, especially if the water is treated by aeration, or in the home or place of business when water is used directly from the source with no treatment. Approximately 41,000 ground water supplies have presently been identified in the United States. Most of these supplies serve small population groups, but at least 66 communities with populations greater than 100,000 persons are served by ground water supplies.

4.6B.2 Description of Process

The ground water supplies often use an aeration process to treat the water. Aeration of water is a mechanical method of removing gases such as carbon dioxide, hydrogen sulfide, methane and odors which may be dissolved or entrained in water. It is also an important method of oxidizing dissolved metals such as iron and manganese.

There are two basic types of aerators--positive (either forced or induced) draft and natural draft. The positive draft, the most efficient type for removal of dissolved gases, is designed to allow water to fall through successive layers of screens or slats to break the water into a fine mist inside a steel, aluminum or redwood chamber. The cascading water is subjected to a high velocity countercurrent of air, either forced or induced. Induced draft aerators use a top mounted blower system which pulls air through the chamber at slightly less than atmospheric pressure. Air passing through the water mist oxidizes the dissolved metals and removes entrained gases. The air then passes through a moisture separator and out the top of the aerator. Forced draft aerators use a blower system to force air from the bottom up through the mist zone and out the top of the aerator chamber.

Figure 4.6B-1. Radon concentrations in water, United States (Du76)
(438 samples; excluding New England)



The natural draft type of aerator uses successive slats or trays to break up falling water into small droplets or a fine mist. These fine droplets are exposed to air which oxidizes the dissolved metals such as iron or manganese.

4.6B.3 Emissions of Radionuclides

All ground water contains some quantity of radon-222 as the result of decay of radium-226, both in the water and the rock and soil matrix surrounding the water. Concentrations of radon-222 measured in public ground water supplies in the United States (except for New England) ranged up to 50,000 pCi/L (Du76). About 75 percent of the supplies fall within a range from nondetectable to 2,000 pCi/L and about 5 percent of the supplies measured exceeded 10,000 pCi/L (figure 4.6B-1).

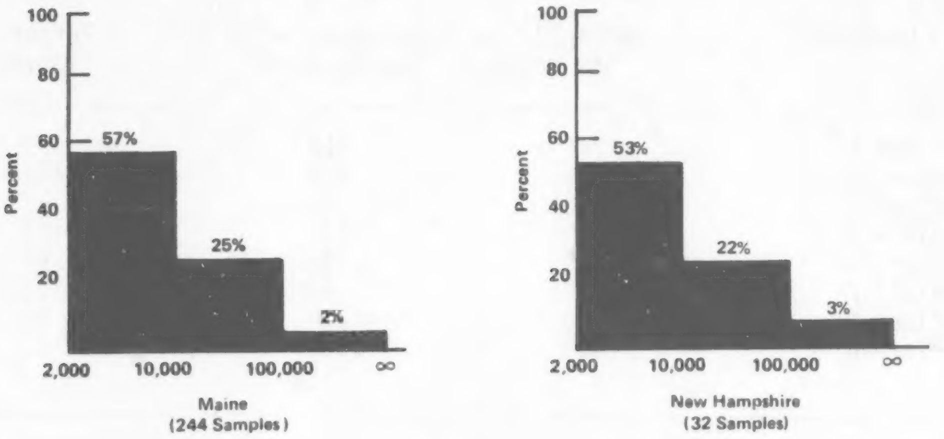


Figure 4.6B-2. Radon concentrations in water, New Hampshire and Maine (Du76)

In Maine and New Hampshire the radon-222 concentrations were much higher than the other areas (Du76) with up to 56 percent of the measured supplies exceeding 2,000 pCi/L and 24 percent exceeding 10,000 pCi/L (figure 4.6B-2).

In order to obtain additional information on radon-222 in drinking water supplies, the EPA has initiated a sampling and measurement program for radon-222 in drinking water supplies in the United States. The initial results of these measurements have been recently reported (EPA79).

Although no direct measurements of radon-222 emissions from ground water treatment plants have been made, the amount of radon-222 released from a plant during the aeration process can be estimated from measurements of the raw and treated water concentrations. Measurements of this type have indicated that a large fraction of the radon-222 is released from the water during aeration (Pa79). Table 4.6B-1 presents data on the amounts of radon-222 removed from water during aeration at 8 water treatment plants.

Table 4.6B-1. Radon-222 removal from water during aeration at water treatment plants (Pa79)

Location	Radon-222 concentration (pCi/L) Raw water	Radon-222 concentration (pCi/L) Treated water	Percent removed
Plant A	746	218	71
Plant B	823	417	49
Plant C	260	70	73
Plant D	225	150	33
Plant E	455	<16	>96
Plant F	235	40	83
Plant G	460	100	78
Plant H	415	160	61

4.6B.4 Model Facility

In order to estimate the potential radioactive emissions and health impacts from ground water treatment plants, a model ground water treatment plant (table 4.6B-2) was developed by assigning the various parameters needed in assessing impacts. The model plant emissions are shown in table 4.6B-3.

Table 4.6B-2. Model ground water treatment plant

Parameter	Value
Plant capacity[a]	1.1E+7 liters/day
Type of treatment[a]	Aeration: lime-soda ash softening
Number of wells[a]	2 deep wells (760 meters) 5 shallow wells (15 meters)
Raw water--	
Radon-222 concentration	1,000 pCi/L
Treated water--	
Radon-222 concentration	150 pCi/L
Radon-222 removal efficiency[a]	0.85

[a] These paramaters were taken from data on the underground water supply serving West Des Moines, Iowa (Pa79).

Table 4.6B-3. Atmospheric emissions of radon-222 from the model underground treatment plant

Radionuclide	Emissions (Ci/y)
Radon-222	3.4

4.6B.5 Health Impact Assessment of the Model Underground Water Treatment Plant

The estimated working level exposures that result from radon-222 emissions from the model underground water treatment plant are listed in table 4.6B-4. These estimates are for a moderate population density suburban type site (Site B, Appendix A).

Table 4.6B-5 estimates the individual lifetime risks and number of fatal cancers resulting from these working level exposures. The lifetime risk of lung cancer to the highest exposed group of individuals is about 1 x 10^{-5}. The lifetime risk of lung cancer to the average individual in the region is estimated to be 3 x 10^{-8}. The number of fatal cancers per year of plant operation is estimated to be 1 x 10^{-3}.

Table 4.6B-4. Working level exposures from radon-222 emissions from model underground watertreatment plant

Source	Maximum individual (WL)	Regional population (person-WL)
Underground water treatment plant	7.8E-6	6.0E-2

Table 4.6B-5. Individual lifetime risks and number of fatal cancers due to radioactive emissions from the model underground water treatment plant

Source	Individual lifetime risk		Expected fatal cancers per year of operation (Fatal cancers)
	Maximum individual	Average individual Region	
Radon-222	~~1.2E+5~~ 1.2E-5	3.4E-8	1.2E-3

REFERENCES

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CHAPTER 5
OTHER SOURCE CATEGORIES

This section includes source categories not within the scope of Chapters 2, 3, and 4--United States Army reactors, United States Navy nuclear shipyards, and particle accelerators. In general, these are minor sources of radioactivity.

5.1 Department of Defense Facilities

5.1A U.S. Army Facilities

5.1A.1 General Description

The U.S. Army Test and Evaluation Command operates two reactors which are used principally for nuclear weapons radiation effects testing of Army and DOD related materiel. One reactor is located at the Army Pulse Radiation Facility (APRF) on the site of the Aberdeen Proving Grounds in Maryland. The second, the Fast Burst Reactor (FBR) is located on the Army's White Sands Missile Range in New Mexico.

In addition to the pulse type reactors, the Army also maintains surveillance at three nuclear power plant reactors that have shut down and been decommissioned. Two of these reactors, the MH-1A and SM-1 are located at Fort Belvoir, Virginia. The other reactor, the SM-1A, is located at Fort Greely, Alaska. Environmental monitoring programs are maintained around these sites to detect and evaluate any significant changes in the radiation levels that may be due to emissions from the reactors.

The Army had a TRIGA MARK F research reactor, operated by the Diamond Ordinance Radiation Facility (DORF), which was shut down for decommissioning in October 1977. This research reactor is located in the Forest Glen section of Walter Reed Army Medical Center in Washington, D.C.

5.1A.2 Process Description

The pulse reactors operated by the Army's Test and Evaluation Command are bare, unreflected, unmoderated, and fueled with enriched uranium. The reactors are capable of self-limiting, super-prompt-critical pulse operations and steady-state operations at low power levels (less than 10 kW). Table 5.1A-1 summarizes the modes of operations of the two reactors.

Forced air cooling systems are used to reduce the temperature of the cores in order to minimize the time between operations. The air exhausted from the reactor buildings is passed through HEPA filters before being released.

Table 5.1A-1. Number and modes of operation of the Army reactor facilities, 1976, (Aa77, De76)

Type of Operation	Number of operations APRF	FBR
Pulse	237	277
Steady state	236	30
Unscheduled terminations	-	11
Total	473	318

5.1A.3 Emissions of Radionuclides

Each reactor operation produces airborne radioactivity in the reactor building due to neutron activation of material in the air and release of fission products from the core. The airborne radioactivity due to fission products from the core is normally small compared to that due to activation. Emissions of radioactive particulates are reduced by the HEPA filters in the exhaust system. The gaseous radioactivity emitted to the atmosphere is principally argon-41 from air activation. Table 5.1A-2 summarizes the emissions from the two sites.

5.1A.4 Health Impact Assessment

No dose estimates were reported for the Fast Burst Reactor at White Sands Missile Range.

At the Aberdeen reactor, gamma doses, from one pulse of 2.0E+17 fissions, were reported for distances of 411 and 1372 meters. The reported doses were 4.7 mrem and 0.008 mrem, respectively.

The maximum annual gamma dose for 1976 was reported to be 500 mrem at a film badge area monitoring station located 1.6 kilometers south of the APRF reactor. This point is within the Aberdeen Proving Ground Military Reservation. A person who continually occupied this location would have a lifetime risk of fatal cancer of 7.0E-3. No population dose assessments were reported for the area surrounding the APRF.

Table 5.1A-2. Emissions of radionuclides from Army pulse reactors, 1976 (Aa77, De76)

Month	APRF Gross beta concentrations[a] (μCi/cm^3)	FBR Gross gaseous activity (Ci)
January	6.3E-14	4.9E-1
February	1.1E-13	1.2
March	1.4E-15	6.3E-1
April	1.4E-15	1.1
May	8.6E-14	1.1
June	3.2E-14	1.2
July	3.6E-14	3.5
August	6.1E-14	9.0E-2
September	3.5E-14	1.3E-1
October	6.0E-14	7.7E-1
November	9.5E-14	8.5E-1
December	1.6E-13	6.1E-1
Total	-	1.2E+1

[a] Only average monthly gross beta concentrations in the reactor building stack were reported. No volumetric air exhaust rate or total emissions were reported.

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5.1B U.S. Navy Facilities

5.1B.1. General Description

The United States Navy had 113 nuclear powered submarines and 11 nuclear powered surface ships in operation at the end of 1978. The ships are powered by pressurized water nuclear reactors. Associated with the operation of these ships are support facilities involved in construction, maintenance, overhaul, and refueling of the nuclear propulsion plants. These facilities include nine shipyards, thirteen tenders, and two submarine bases.

5.1B.2 Process Description

Shipboard nuclear reactors primarily release small amounts of radioactivity in liquid discharges. However, less than 100 curies per year of carbon-14 is released into the atmosphere from U. S. naval nuclear powered ships and their supporting facilities. Most of the carbon-14 is released at sea, over twelve miles from shore. The naval nuclear reactors and their support facilities involved with handling radioactive materials have air exhaust systems equipped with HEPA filters to reduce emissions. The exhausted air is monitored during discharge. The concentrations of radioactivity in air exhausts were below those levels normally present in the atmosphere.

5.1B.3 Emissions of Radionuclides

Table 5.1B-1 summarizes estimates of airborne emissions (Mi79) from a typical nuclear naval shipyard. These estimates, used in environmental pathway analysis, are higher than any measurements made in the past five years from any shipyard.

5.1B.4 Health Impact Assessment of a Typical Naval Shipyard

Table 5.1B-2 summarizes the annual total body doses to the maximum individual and to the population within 80 kilometers of a typical nuclear naval shipyard. The individual dose estimates were based on the estimated releases shown in table 5.1B-1. The population dose was based on both liquid and airborne emissions. The individual lifetime risks and health effects from the operation of a nuclear naval shipyard are summarized in table 5.1B-3. Assuming all nine shipyards have similar health risks, the total number of fatal cancers from radiological work at all naval shipyards is estimated to be less than 1.8E-3 for each year of operation.

Table 5.1B-1. Estimated atmospheric emissions of radionuclides from a typical nuclear naval shipyard (Mi79)

Radionuclide	Emissions (Ci/y)
Argon-41	4.1E-1
Cobalt-60	1.0E-3
Tritium	1.0E-3
Carbon-14	1.0E-1
Krypton-83m	2.0E-2
Krypton-85m	2.4E-2
Krypton-85	1.0E-3
Krypton-87	5.0E-2
Krypton-88	2.0E-2
Xenon-131	5.0E-3
Xenon-133m	1.0E-2
Xenon-133	2.1E-1
Xenon-135	2.5E-1

Table 5.1B-2. Annual radiation doses from a typical nuclear naval shipyard (Mi79)

Organ	Maximum individual (mrem/y)	Population[a] (person-rem/y)
Total body	<2.0	<1.0

[a] To the population within 80 km.

Table 5.1B-3. Individual lifetime risks and number of fatal cancers from radioactive emissions at a typical nuclear naval shipyard

Source	Individual lifetime risks Maximum individual	Average individual	Expected fatal cancers[a] per year of operation (Fatal cancers)
Nuclear shipyard	<2.8E-5	<1.4E-8	<2.0E-4

[a] To the population within 80 km.

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5.2 Particle Accelerator Facilities

5.2.1 General Description

Accelerators are devices for imparting high kinetic energy to electrons or positively charged particles (such as alpha particles, protons, and deuterons) by electrical or magnetic fields. In a typical operation, the accelerated particles travel in an evacuted tube or enclosure. The particles impinge on a metallic or gaseous target, producing secondary radiation.

There are three basic accelerator designs categorized according to the means used to achieve the particle velocity: (1) constant direct current (DC) field machines, (2) incremental acceleration machines, and (3) magnetic field accelerators.

The constant DC field machines, sometimes called "Potential-drop" accelerators, operate at very high voltages. Charged particles achieve greater kinetic energy as they are accelerated through the direct current electric field toward the target. These accelerators are named according to the type of power supply used to generate the high DC voltage they require. The principal design types in this class are the Van de Graaff, Cockcroft-Walton, Dynamitron, Resonant Transformer and Insulating Core Transformer.

Incremental accelerators accelerate particles by temporally varying electric fields which impart kinetic energy in discrete increments. Therefore, the velocity of the particle increases stepwise rather than in a continuous manner. The principal design types of this class are the linear accelerator (linac) and the cyclotron.

The betatron is the only type of magnetic field accelerator. Electrons are continuously accelerated to desired energy levels in the betatron by a temporal variation in the intensity of a magnetic field.

Accelerators have a variety of applications. Potential-drop machines are commonly used for radiography, activation analysis, food sterilization and preservation, industrial processing (ion implantation), radiation therapy, and research. Electron linear accelerators are widely used in research by universities, government, industry, and medical laboratories. Accelerators have become increasingly popular in medical radiation therapy as replacements for teletherapy sources (e.g., cobalt-60). Cyclotrons, which accelerate the heavier positively charged particles, are commonly used in physics laboratories and in the radiopharmaceutical industry. The betatron has many of the same applications as the electron linacs and electron accelerating potential-drop accelerators.

Estimates of the number of particle accelerators in the United States were published by the Bureau of Radiological Health (BRH78). In 1977, over 1100 accelerators were reported in use in this country, not including Federally-owned accelerators. Most of the very high energy physics research accelerators are owned by the Department of Energy and are briefly discussed in Chapter 3.

Linear accelerators are the most widely used machines in the United States; about 70 percent of the linacs are used in medical applications. The percent, by type of machine, of the total number of accelerators reported is as follows: linacs, 50 percent; Van de Graaff, 15 percent; Neutron Generators, 17 percent; Resonant and Insulating Core Transformers, 6 percent; Cyclotrons, 3 percent; Betatrons, 6 percent; Cockcroft-Walton (ion implanters), 3 percent.

5.2.2 Process Description

At research facilities there is diversity in operational modes. The size or energy of the accelerators, the type of particle accelerated, and the target material used are important variables which make it difficult to predict airborne effluent composition over any extended time period. Also, there are a large number of radionuclides that may be present in airborne emissions as a result of nuclide production at medical or radiopharmeceutical facilities (Section 2.3 of this report).

Possible sources of airborne contamination at accelerator facilities include: (1) loss of target integrity, (2) handling of targets in laboratory hoods and glove boxes, and (3) activation of dust and room air.

Loss of target integrity may involve rupture of powder targets, flaking-off of the activated surface layer of solid targets, or leakage of flow-type targets. The vacuum pump exhaust may be an especially important pathway for gaseous radionuclides, particularly tritium. Potential airborne radioactive contaminants from a loss of target integrity depend on such factors as target composition, particle acceleration, energy, and time of irradiation.

Although not directly related to the actual operation of the accelerator, radiochemical operations in preparing targets for use in the accelerator and in handling exposed targets in laboratory hoods and glove boxes are potential sources of airborne radionuclides at an accelerator facility.

Interaction of primary and secondary particles with dust in the air is another potential source of emissions of radioactive materials from accelerator facilities. This source may be minimized by good housekeeping procedures in the accelerator room and by filtration of the ventilation air at either intake or exhaust or both. Another source of radioactive dust particles is nuclear interaction with the structural material of the accelerator. Welding, soldering, and working with accelerator parts may initiate radioactive dust and vapors in the air presenting a potential internal contamination problem for occupational exposures and probably contribute little radioactivity to the environment.

If the primary particle beam emerges into the air before striking the target and has sufficient energy, nuclear reactions with gases in the air will occur. This interaction of the primary and secondary particles with air in the accelerator hall will vary depending upon the hall size, ventilation rate, type of particle accelerated, energy, and the nature of the target. Protons and heavy ions accelerated to high energies produce nuclear reactions directly. All energetic accelerated particles give rise to protons and neutrons as secondary radiations from interaction in various targets. Table 5.2-1 lists some radionuclides and their half-lives that are associated with dust and air activation at accelerator facilities.

5.2.3 Emissions of Radionuclides

Emissions of radioactive materials at accelerator facilities are principally dependent on production rates of air activation products which are functions of the following factors:

1) The incident flux or secondary flux produced by the accelerator;
2) The density of the target gas atoms in the beam path;
3) The appropriate cross-section for the reaction at the interaction energy;
4) The buildup factor, which is based on exposure time. Its maximum is at the equilibrium condition;
5) The decay that occurs between termination of the irradiation and the time of exposure to the radionuclides.

The constant field machine delivers a beam in the energy range of 1 to 19 MeV, which is, for the most part, below the threshold energies for air activation shown in table 5.2-1. Gaseous tritium, however, can be released from these machines when they are used to produce neutrons. Neutrons are generated by bombarding a

Table 5.2-1. Half-lives of activation radionuclides produced in accelerator facilities

Isotope	Half-life
Oxygen-15	2 min
Nitrogen-13	10 min
Nitrogen-16	7 s
Oxygen-14	1 min
Carbon-11	20 min
Argon-41	1.9 h
Beryllium-7	53 d
Tritium	12 y
Sulfur-38	37 min
Fluorine-18	2 h

tritium target with deuterium. During this process tritium may be knocked off the target and released to the room or atmosphere through the vacuum pump exhaust--about 100 to 300 mCi per target. Typical targets contain about 5 Ci of tritium. The small neutron flux produced (about 50 n/cm^2/s) is not considered large enough to produce significant activation products.

Of the incrementally accelerated machines, the most popular is the 18 MeV medical electron linac. At this energy the photons produced are usually below the threshold energies for air activation shown in table 5.2-2. However, it is possible to use the measured thermal neutron flux around an 18 MeV electron linac to calculate the amount of argon-41 and carbon-14 produced in the air. The flux was reported to be about 440 n/cm^2/s. Assuming 2000 hours of operation per year, the activity produced in a room 27 m^3 is 1.0E-4 and 1.0E-9 curies, respectively.

Measurements at a 40 MeV linac detected trace amounts of chlorine-39 and argon-41. The common air activation products, carbon-11, nitrogen-13, and oxygen-15, were also found to be produced at rates that ranged from <0.1 µCi per pulse of 20 MeV electrons without a bremsstrahlung target to 2 µCi per pulse of 45 MeV electrons with a bremsstrahlung converter. The total facility release quantities were reported to be in the curie range.

Production rates of oxygen-15, nitrogen-13, and carbon-11 in the presence of a high energy neutron flux at a cyclotron facility have been estimated. The conditions assumed were a 1 µA proton beam striking a thick beryllium target at 100 MeV. This arrangement produces neutrons at a rate of of 2.0E+12 n/sec with energies distributed around 100 MeV. Assuming a 1 cm^2 beam traversing 1 meter of air, and using the abundance of carbon, nitrogen, and oxygen isotopes in air, production rates were calculated (table 5.2-3). Decay during production was not considered since activation products are free to be exhausted immediately after production. An estimate of the thermal neutron flux is not available for this situation; thus, production rates of argon-41 and other (n,x) products cannot be calculated.

Table 5.2-4 lists the estimated annual emissions from typical particle accelerator facilities.

Table 5.2-2. Nuclear reactions responsible for some airborne radioactivity

Reaction	Parent nuclide	Isotope produced	Threshold energy (MeV)
(γ,n)	Nitrogen-14	Nitrogen-13	10.5
(γ,n)	Oxygen-16	Oxygen-15	15.7
(γ,n)	Carbon-12	Carbon-11	18.7
(n,2n)	Nitrogen-14	Nitrogen-13	11.3
(n,2n)	Oxygen-16	Oxygen-15	18.0
(n,2n)	Carbon-12	Carbon-11	20.0
(n,pn)	Oxygen-16	Oxygen-15	10.0
	Nitrogen-14	Nitrogen-14	10.0
(n,γ)	Argon-40	Argon-41	NA

NA Not applicable.

Table 5.2-3. Production rates of oxygen-15, nitrogen-13, and carbon-11 at a 100 MeV cyclotron producing 2.0E+12 neutrons per second

Reaction	Production rate (atoms/s)	Amount produced in 4 hours (Ci)
$^{12}C(n,2n)^{11}C$	6.0E+4	7.0E-5
$^{14}N(n,2n)^{13}N$	5.0E+7	5.3E-2
$^{16}O(n,2n)^{15}O$	9.0E+7	9.4E-2

The most popular accelerator is of the incremental or cyclic type. Half of the accelerators in the United States are linacs, 90 percent of which are electron linacs in the energy range of 1 to 19 MeV and 1 to 10 kilowatts in power. These characteristics are typical of cancer therapy machines widely used in hospitals. For this reason the 18 MeV medical linac is considered to be a typical cyclic accelerator.

The betatron is a unique accelerator class. However, its radiation characteristics and applications are similar to electron linacs. The production of airborne radioactivity by the typical linac can also be applied to the betatron. Therefore, a typical magnetic field machine will not be discussed.

5.2.4 Typical Facility

The information for a typical facility is generalized to the various classes of accelerators. Figure 5.2-1 shows the range of operating characteristics for various types of accelerators. Although accelerators vary within a given class, some generalizations can be made.

Of the constant potential type machines, the Van de Graaff accelerators are the most prevalent. About 15 percent of all registered accelerators in 1977 belonged in this category. Furthermore, 78 percent of the Van de Graaff accelerators were in the energy range of 1 to 19 MeV with intensity ranging from 1 μA to 1 mA. To estimate health impacts, a 6 MeV Van de Graaff using a tritium target is considered as typical of a constant potential type of machine.

The other type of cyclic accelerator, the cyclotron, generally operates at higher energies and power levels than other machines in use and therefore has the potential of producing greater quantities of airborne raidoactivity. A typical research cyclotron accelerates ions at about 100 MeV and at very low currents into scattering targets to produce secondary radiation. The secondary radiation is in the form of hard gamma rays (>10 MeV). These gammas are presumably responsible for producing nitrogen-13, oxygen-15, and carbon-11. At lower energies, cyclotrons would probably not produce secondary radiation with sufficient energy to produce even nitrogen-13 via the (γ,n) reaction. Although increased research is being performed in the area of neutron production for medical applications, it is difficult to estimate the annual utilization of a cyclotron for neutron beam generation. For typical cyclotron facilities, then, it will be assumed that 2 percent of the yearly operating time (1000 hours) is allocated to neutron production.

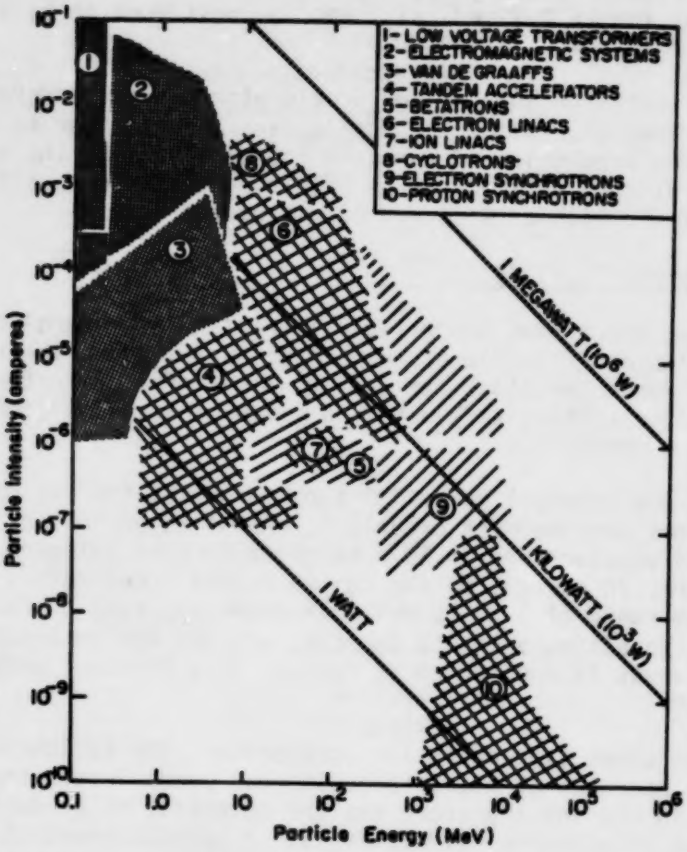

Figure 5.2-1. Operating characteristics of accelerators. (Particle beam intensity versus particle energy, for several types of accelerators. The dark-toned areas relate to electron and ion accelerators; middle tones to ion accelerators only; light tones to electron accelerators only.)

Table 5.2-4 lists the estimated annual emissions from typical accelerators. Table 5.2-5 summarizes the characteristics of a typical accelerator facility.

Table 5.2-4. Estimated annual emissions from typical particle accelerators (Te79)

Radio-nuclide	100 MeV Cyclotron (Ci)	18 MeV Electron Linac (Ci)	6 MeV Van de Graaf[a] (Ci)
Carbon-11	2.0E-3		
Nitrogen-13	4.0E-2		
Oxygen-15	1.0		
Tritium			1
Carbon-14		1.0E-9	
Argon-41		1.0E-4	

[a] Tritium target used for neutron generation; release estimates include emissions from laboratory hoods due to tritium target handling operations.

Table 5.2-5. Typical accelerator facility

Parameter	Value
Type of accelerator:	6 MeV Van de Graaff with tritium target--operating 3000 h/y
	18 MeV electron linac operating 2000 h/y
	100 MeV research cyclotron operating 1000 h/y
Emissions control:	None
Stack characteristics:	
Height	16.8 meters (roof type)

5.2.5 Health Impact Assessment of Typical Accelerators

The diverse operational modes practiced by facilities using accelerators make it difficult to predict airborne effluent compositions over any extended time period. Even at medical accelerators, routine changes in radionuclide production and target characteristics make it difficult to predict realistic source term values. However, table 5.2-6 represents estimates of the annual radiation doses resulting from radioactive emissions from typical accelerators. These estimates are for a site in the suburbs of a large Midwestern city (Site B, Appendix A). The nearest resident was assumed to live 1 kilometer from the site.

Individual fatal cancer risks and committed fatal cancers to the population within 80 km are estimated in table 5.2-7.

Table 5.2-6. Annual radiation doses due to radioactive emissions from typical accelerators

Type of accelerator	Maximum individual (mrem/y)	Average individual (mrem/y)	Population (person-rem/y)
6 MeV Van de Graaf	1.1E-4	2.4E-07	5.9E-4
18 MeV Electron linac	4.2E-8	1.3E-10	3.1E-7
100 MeV Research cyclotron	9.6E-5	2.1E-09	5.1E-6

Table 5.2-7. Individual lifetime risks and number of fatal cancers due to radioactive emissions from typical accelerators

Type of accelerator	Individual lifetime risks Maximum individual	Average individual	Expected fatal cancers per year of operations (Fatal cancers)
6 MeV Van de Graaf	1.6E-09	3.4E-12	1.2E-07
18 MeV Electron linac	5.9E-13	1.8E-15	6.4E-11
100 MeV Research Cyclotron	1.3E-09	2.9E-14	1.0E-09

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APPENDICES

APPENDIX A
ASSESSMENT METHODOLOGY

A.1 Introduction

The general methodology used in the generic assessments presented in this report consisted of the following parts:

1) a description of a model or typical facility for the source category,

2) a choice of one or more generic sites appropriate to the source category,

3) an assignment of a source term (Ci/y) and source related quantities (e.g., release height, plume rise),

4) a calculation of the individual, average, and collective doses due to air immersion, ground surface exposure, inhalation, and ingestion of radionuclides,

5) a health risk assessment based on the doses to the various organs and the working-level exposure from radon-222.

Assumptions made at each step were based on being realistic but not underestimating the impact of a release. The following sections describe these steps in more detail. (See Appendix B for health risk assessment details.)

A.2 Model/Typical Facility

For each source category, a representative facility was designated. In some instances (e.g., nuclear power plants), extensive information was available on release rates and source considerations influencing dispersion (e.g., release height and exit velocity). In such cases, a model facility was designed to represent an average facility for the source category. For other source categories (e.g., radiopharmaceutical industry), industry wide information was sparse. In these cases, data for a particular facility considered representative of the source category was used for the assessment and the facility was identified as a typical facility.

A.3 Generic Sites

Six generic sites were characterized for the purpose of assessing different source categories. These sites were chosen by first identifying locations of facilities within each source category and then identifying a few of them which typified the types of locations where such facilities might be located. Factors which entered into this judgment included geographic location, population density, and food crop production.

On the basis of similarities between representative sites for the different source categories, six generic sites (designated A, B, C, D, E, and F) were chosen which were believed to adequately represent potential sites for all of the source categories considered. For some source categories, one site was sufficient (e.g., uranium mining) while others required as many as four sites to represent the source category (e.g. fossil fuel power plants). While the data used to characterize the generic sites were obtained for specific locations, there would not necessarily be a facility at that location for any specific source category.

Sites A and B represent urban locations. Site A characterizes a very large metropolitan city: the maximum case with respect to population density and overall population within 80 km (New York City, New York). Site B represents the near suburbs of a large Midwest city (St. Louis, Missouri). Site C was selected to depict the phosphate industry since this location has a heavy concentration of phosphate mining and milling (Polk County, Florida, near Bartow). Site D represents a rural setting in the central portion of the United States (near Little Rock, Arkansas). Site E exhibits the characteristics associated with the uranium industry and other mining endeavors (Grants, New Mexico). Site F is a remote, sparsely populated location in the Northwest which represents a minimal impact on the general population (near Billings, Montana). Sites C through F are considered rural locations. Table A-1 gives the important characteristics of these generic sites.

A.4 Source Characterization

Sources were characterized by the release rate (Ci/year) of each emitted radionuclide. Radionuclides released as particulates or reactive vapors were assigned a deposition velocity V_d of 1 cm/s unless otherwise indicated and a precipitation scavenging factor of approximately 1E-5 C s^{-1} where C is the average precipitation rate in m/year. See table A-3 for the actual values used for the generic sites. An effective release height was assigned to each source based on the release height and any expected plume rise. In general, no credit was given for plume rise unless it was clearly indicated. Because the depletion calculation method in AIRDOS-II required very long run times for low level releases, ground level releases were given an effective release height of 10 m.

Table A-1. Characteristics of the generic sites

Site A--New York

Meteorological data:	New York/LaGuardia (WBAN=14732)
Stability Categories:	A-F
Period of Record:	65/01-70/12
Annual Rainfall:	102 cm
Average Mixing Height:	1000 m
Population (0-8 km):	9.23E+5 persons
(0-80 km):	1.71E+7 persons
Dairy Cattle (0-80 km):	1.72E+5 head
Beef Cattle (0-80 km):	1.17E+5 head
Vegetable Crop Area: (0-80 km)	3.77E+4 ha

Site B--Missouri

Meteorological data:	St. Louis/Lambert (WBAN=13994)
Stability Categories:	A-G
Period of Record:	60/01-64/12
Annual Rainfall:	102 cm
Average Mixing Height:	600 m
Population: (0-8 km):	1.34E+4 persons
(0-80 km):	2.49E+6 persons
Dairy Cattle (0-80 km):	3.80E+4 head
Beef Cattle (0-80 km):	6.90E+5 head
Vegetable Food Crop Area: (0-80 km)	1.64E+4 ha

Table A-1. Characteristics of the generic sites--continued

Site C--Florida

Meteorological data:	Orlando/Jet Port (WBAN=12815)
Stability Categories:	A-E
Period of Record:	74/01-74/12
Annual Rainfall:	142 cm
Average Mixing Height:	1000 m
Population: (0-8 km):	6.67E+3 persons
(0-80 km):	1.51E+6 persons
Dairy Cattle (0-80 km):	2.76E+4 head
Beef Cattle (0-80 km):	2.57E+5 head
Vegetable Crop Area: (0-80 km)	1.39E+4 ha

Site D--Arkansas

Meteorological data:	Little Rock/Adams (WBAN=13963)
Stability Categories:	A-F
Period of Record:	72/02-73/02
Annual Rainfall:	127 cm
Average Mixing Height:	600 m
Population: (0-8 km):	1.18E+4 persons
(0-80 km):	5.92E+5 persons
Dairy Cattle (0-80 km):	1.19E+4 head
Beef Cattle (0-80 km):	2.57E+5 head
Vegetable Crop Area: (0-80 km)	2.94E+3 ha

Table A-1. Characteristics of the generic sites--continued

Site E--New Mexico	
Meteorological data:	Grants/Gnt-Milan (WBAN=93057)
Stability Categories:	A-F
Period of Record:	54/01-54/12
Annual Rainfall:	20 cm
Average Mixing Height:	800 m
Population: (0-8 km):	0 persons
(0-80 km):	3.60E+4 persons
Dairy Cattle (0-80 km):	2.30E+3 head
Beef Cattle (0-80 km):	8.31E+4 head
Vegetable Crop Area: (0-80 km)	2.78E+3 ha

Site F--Montana	
Meteorological data:	Billings/Logan (WBAN=24033)
Stability Categories:	A-F
Period of Record:	67/01-71/12
Annual Rainfall:	20 cm
Average Mixing Height:	700 m
Population: (0-8 km):	0 persons
(0-80 km):	1.19E+4 persons
Dairy Cattle (0-80 km):	1.86E+3 head
Beef Cattle (0-80 km):	1.47E+5 head
Vegetable Crop Area: (0-80 km)	1.77E+4 ha

A.5 Environmental Pathway Modeling Computer Program

AIRDOS-II (Mo77) was used to calculate the individual and collective doses and working level exposures for these assessments. The program was modified to assess area sources using the methodology described by Culkowski and Patterson (Cu76). Working levels associated with radon-222 were calculated on the assumption of a 70 percent equilibrium, a value considered representative of indoor exposure conditions (Ge78).

Air concentrations are ground level sector averages. Dispersion is calculated from annual average meteorological data. Depletion due to dry deposition and precipitation scavenging was calculated for particulates and reactive vapors. AIRDOS-II provides no means for calculating resuspended air concentrations or subsequent redeposition to the ground surface. The air concentrations are the basis for inhalation and submersion dose calculations.

Ground surface concentrations are calculated on the basis of the deposition rate due to dry deposition and precipitation scavenging. A 100-year accumulation period was generally used unless otherwise indicated. Radiological decay was the only process considered to deplete the ground surface concentration. Effective decay constants were used to calculate the concentrations at the end of the accumulation period for nuclides which are members of decay chains.

Ingestion rates were calculated using the TERMOD portion of AIRDOS-II. Initial assessments indicated that transfers from the soil pool were often unrealistically high, especially for the long-lived radionuclides. This result was due in part to the equilibrium (rather than 100-year) values calculated by TERMOD and in part to the TERMOD element dependent parameters (Ki76). Since no satisfactory resolution of these difficulties was practicable, the decision was made to consider only those transfers associated with foliar deposition on food and forage. For selected nuclides, the calculated ingestion intake rates are a low estimate of the values after 100 years' accumulation. However, this approach is not expected to significantly underestimate the overall risk from the source categories considered and avoids some extreme overestimates that would otherwise have been made.

A.6 Individual Dose Assessment

The maximum individual was assessed on the following basis:

1) The maximum individual for each source category is intended to represent an average of the individuals living near each facility within the source category. The individual was assumed to be located approximately 500 meters from the point of release in the predominant wind direction. For area sources this location was nominally 500 meters from the edge of the source.

2) The organ dose-equivalent rates in the tables are based on the calculated environmental concentrations. For inhaled or ingested radionuclides, the dose-equivalent rates are actually the 50-year committed dose-equivalent rates, i.e., the internal dose-equivalents which would be delivered up to 50 years after an intake. The individual dose equivalent rates in the tables are in units of mrem/y.

3) Since the risk assessment is based on an entire lifetime spent at the calculated environmental concentrations, an adult model was appropriate for dosimetry.

4) The individual is assumed to home-grow a portion of his or her diet consistent with the type of site. Individuals living in urban areas were assumed to consume much less home produced food than an individual living in a rural area. The fractions of home produced food consumed by individuals for the generic sites are shown in table A-2. For Site B the portion of the the individual's diet that was not locally produced was assumed to come from the average of the assessment area. Subsequent trial runs showed little difference between assuming that the balance of the maximum individual's diet comes from the assessment area or that it is imported from outside the assessment area. Some assessments for Site B were performed using the rural food source fractions for Sites C-F in table A-2. These are identified in the text as Site B with rural food source fractions.

A.7 Average Individual

Dose rates and working level exposures for an average individual within 80 km of a source were obtained by dividing the collective dose rates and working level exposures for the region (see A.8) by the population of the region.

Table A-2. Sources of food for the maximum individual

Food	Site A			Site B			Sites C-F		
	F1	F2	F3	F1	F2	F3	F1	F2	F3
Vegetables	7.6	0.	92.4	7.6	92.4	0.	70.0	30.0	0.
Meat	.8	0.	99.2	.8	99.2	0.	44.2	55.8	0.
Milk	0.	0.	100.	0.	100.	0.	39.9	60.1	0.

F1 and F2 are the percentages produced at the individual's location and within the 80 km assessment area, respectively. The balance of the diet, F3, is considered to be imported from outside the assessment area with negligible radionuclide concentrations due to the assessed source.

A.8 Collective Dose Assessment

The collective dose assessment to the population within an 80 km radius of the facility under consideration was performed as follows:

1) The population distribution around the generic site was based on the 1970 census. The population was assumed to remain constant in time.

2) Average agricultural production data for the State in which the generic site is located were assumed for all distances greater than 400 meters from the source. For distances less than 400 meters no agricultural production takes place.

3) The population in the assessment area consumes food from the assessment area to the extent that the calculated production allows. Any additional food required is assumed to be imported without contamination by the assessment source. Any surplus is not considered in the assessment.

4) The collective organ dose-equivalent rates are based on the calculated environmental concentrations. Fifty-year dose commitment factors (as for the individual case) are used for

ingestion and inhalation. The collective dose equivalent rates in the tables can be considered to be either the dose commitment rates after 100 years of plant operation or equivalently the doses which will become committed for up to 100 years from the time of release from one year of plant operation.

A.9 AIRDOS-II Parameters and Input Data

A sample computer input for the AIRDOS-II code as used in these assessments (table A-4) is annotated to enable the reader to understand the input format. Moore (Mo77) supplies a detailed explanation of the code.

Mixing Height and Scavenging

Table A-3 summarizes the mixing heights, rainfall rates, and scavenging coefficients used for the generic sites. A dry deposition velocity of .01 m/s was used for particulates and reactive vapors (e.g., elemental iodine) unless otherwise indicated.

Table A-3. Some site descriptors used with AIRDOS-II

Generic site	Average mixing height (m)	Rainfall rate (cm/y)	Scavenging coefficient (s^{-1})
Site A	1000	102	1.0E-5
Site B	600	102	1.0E-5
Site C	1000	142	1.0E-5
Site D	600	127	1.0E-5
Site E	800	20	2.0E-6
Site F	700	20	2.0E-6

The average mixing height is the distance between the ground surface and a stable layer of air where no further mixing occurs. This average was computed by determining the harmonic mean of the morning mixing height and the afternoon mixing height for the location (Ho72). The rainfall rate (USGS70) determines the value used for the scavenging coefficient. No attempt was made to be more accurate than one significant figure for both average mixing height and scavenging coefficient. Sites E and F are relatively dry locations as reflected by the scavenging coefficients.

Meteorological Data

To demonstrate the methodology of converting from a standard format for meteorological data (STAR format) to the AIRDOS-II format, tables A-5 and A-6 are included. Site B meteorological data are presented as an example in table A-5. A utility Fortran IV program (Mo78) converts the STAR data to meet the input requirements of the AIRDOS-II code (see lines 1200-4200 of table A-4).

STAR (an acronym for STability ARray) meteorological data summaries were obtained from the National Climatic Center, Asheville, North Carolina. Data for the station considered most representative for each generic site were used. Generally, these data are from a nearby airport. The station used is identified by the corresponding WBAN number in table A-1.

Dairy and Beef Cattle

Dairy and beef cattle distributions are part of the AIRDOS-II input. A constant cattle density is assumed except for the area closest to the source or stack in the case of a point source, i.e., no cattle wthin 400 m of the source. The cattle densities are provided by State in table A-7. These densities were derived from data developed by NRC (NRC75). Milk production density in units of liters/day-square mile was converted to number of dairy cattle / square kilometer by assuming a milk production rate of 11.0 liters/day per dairy cow. Meat production density in units of kilograms/day-square mile was changed to an equivalent number of beef cattle/square kilometer by assuming a slaughter rate of .00381 day^{-1} and 200 kilograms of beef/animal slaughtered. A 180-day grazing period was assumed for dairy and beef cattle.

Population

The population data for each generic site were generated by a computer program (At74) which utilizes an edited and compressed version of the 1970 United States Census Bureau's "Master Enumeration District List with Coordinates" containing housing and population counts for each census enumeration district (CED) and the geographic coordinates of the population centroid for the district. In the Standard Metropolitan Statistical Areas the CED is usually a "block group" which consists of a physical city block. In other areas the district used is called the "enumeration district," and it may cover several square miles in a rural area.

There are approximately 250,000 CEDs in the United States and the average population is about 800. The position of the population centroid for each CED was marked on the district maps by the individual census official responsible for each district and is based only on his judgment from inspection of the population distribution on the map. The CED entries are sorted in ascending order by longitude on the final data tape.

The resolution of a calculated population distribution cannot be better than the distribution of the CEDs. In a metropolitan area the resolution is often as small as one block, but in rural areas it may be on the order of a mile or more.

Vegetable Crop Area

A certain fraction of the land within 80 km of the source is used for vegetable crop production which is assumed to be uniformly distributed throughout the entire assessment area with the exception of the first 400 meters from the source. Information on the vegetable production density in terms of kilograms (fresh weight)/day-square mile were obtained from NRC data (NRC75). The vegetable crop fractions (table A-7) by State were obtained from the production densities by assuming a production rate of 2 kilograms (fresh weight)/year-square meter (NRC77).

Food Intake

Referring to the sample input table (table A-4), lines 21800-21900 determine what percentage of milk, beef and vegetables is produced and consumed at each environmental location; is produced throughout the assessment area (average value for the whole assessment area) and consumed locally; and is produced outside the assessment area (imported) and consumed locally. The imported fraction is assumed to have no radioactivity content.

Table A-2 summarizes the ingestion values used for each generic site for the maximum individual. These values are based on a USDA report (USDA72). Sites A and B utilize data on urban locations while rural sites are based on rural-farm situations. The F1 ratios are obtained by dividing the home-produced quantity by the quantity from all sources. The beef ratios are actually meat values, i.e., beef and pork. The vegetable ratios only include fresh vegetables.

For population exposure estimates, the AIRDOS-II code determines the imported fraction needed to supply the nutritional requirements of the entire population within 80 km. The quantity of food that is not imported is assumed to be grown or produced throughout the entire assessment area and consumed by the population within the assessment area as an average value for the entire assessment area. For a site that produces more food than is needed for the population within the assessment area, this food is assumed to be exported outside the assessment area. No collective doses were calculated for such exported foods.

The ingestion pathway is calculated by the TERMOD portion (Ki76) of AIRDOS-II. The input values shown in table A-8 were used and are independent of location. The soil transfer parameters, TAURG and TAUSP, were set to zero since only contamination from foliar deposition was considered.

Internal Doses

Internal doses to each organ were calculated for inhalation and ingestion using dose conversion factors from table A-9. Sources for the dose conversion factors in table A-9 are listed in table A-10. The choice of organs to be considered was dictated by AIRDOS-II. The dose to the bone is an average dose to hard bone, not bone marrow. The G.I. tract dose is the dose delivered to the lower large intestine. The total body dose is generally a mass weighted mean of the dose to a number of specific organs. Where dose conversion factors for particular organs are not shown in table A-9, the total-body factor is used as a default value.

Doses to other soft tissue used for risk assessments (see Appendix B) were estimated by the muscle dose values. In those cases where the muscle dose conversion factor is actually a total body dose conversion factor which included bone as one of the specific organs, the muscle dose may be overestimated.

Inhalation dose conversion factors for uranium-235 and members of the uranium and thorium series were calculated on the basis of the Task Group Lung Model (TGLD66, ICRP72) for clearance class Y and an effective particle size (AMAD) of 1.0 micrometer. Inhalation dose conversion factors for strontium-90 and technetium-99 were similarly calculated but for clearance classes D and W respectively. Details of the calculational models and assumptions for other nuclides may be found in the references in table A-10.

Working level exposures associated with radon-222 were calculated assuming an indoor exposure at 70 percent equilibrium (Ge78) (i.e., 100 pCi/L radon-222 = 0.7 working level). Accordingly, no inhalation doses were calculated for radon-222 or its short-lived progeny.

TERMOD Input Data

Element dependent input parameters for the TERMOD calculations were taken from Killough (Ki76, Table 2-7). The value of f_m for radium was corrected from 1.5E+2 to 1.5E-2.

External Doses

External doses to each organ were calculated for air submersion and ground surface exposure using the appropriate dose conversion factors from table A-9. Air submersion doses are for a semi-infinte exposure to the ground level air concentration. Ground surface exposures are for an infinite plane with the ground surface concentration at the particular location. Noble gases (e.g., xenon and krypton) were assumed to be nondepositing. Therefore, even through table A-9 indicates surface DCFs for some noble gases there were no surface doses from these radionuclides.

Table A-4. Sample input for AIRDOS-II

```
100    OPTIONS  1 2
200             1        16         2        15
300     0.0        0.0        0.0        0.0        0.0        0.0        0.0
400    AREA                                                                                    300: Specific plume rise
500    8000.0                                                                                    for each Pasquill
600          265        503        700       1007       1623       2517       3501               stability class
700         7002      10070      16230      25170      35010      50340      70020      5034   600-800: Distances from
800                                                                                              stack in meters
900    AIR                                                                                       (circular option)
1000         600
1100   40.0      282.0      .0728      .1090      .1455                                        1000: Mean mixing height
1200   0.1100.0840.0790.0610.0420.0350.0390.0380.0410.0480.0750.1010.0790.0610.0540.053          in meters
1300    1.63 1.85 1.30 1.45 1.35 1.33 1.45 1.23 1.32 1.58 1.64 1.45 1.38 1.54 1.50 1.67
1400    1.86 2.30 1.82 1.63 1.85 1.80 1.88 1.83 1.87 1.81 1.84 1.99 1.63 2.06 2.25 1.98
1500    3.55 3.40 2.88 2.80 3.15 2.81 2.73 3.07 3.09 3.24 3.38 3.15 2.88 3.38 3.35 3.81
1600    5.03 5.01 4.35 3.86 3.53 3.23 3.17 3.78 4.26 4.71 5.11 5.54 4.47 4.43 4.29 4.72
1700    3.83 3.67 3.29 3.29 3.14 3.07 2.93 3.19 3.54 3.55 3.61 3.69 3.55 3.61 3.57 3.69
1800    1.88 1.85 1.75 1.75 1.46 1.61 1.64 1.88 1.86 1.76 1.74 1.91 1.86 1.88 1.99 1.99
1900    0.77 0.77 0.77 0.77 0.77 0.77 0.77 0.77 0.77 0.77 0.77 0.77 0.77 0.77 0.77 0.77
2000    2.12 2.27 1.82 1.97 1.87 1.85 1.97 1.73 1.84 2.09 2.14 1.97 1.91 2.06 2.02 2.15
2100    2.70 3.06 2.66 2.56 2.74 2.60 2.62 2.71 2.71 2.72 2.78 2.75 2.54 2.84 3.06 2.80
2200    4.40 4.25 3.83 3.72 3.85 3.78 3.82 3.80 3.97 4.13 4.26 4.15 3.93 4.18 4.17 4.45
2300    5.97 5.98 5.43 5.02 4.69 4.39 4.43 4.98 5.40 5.73 6.31 6.88 5.95 5.67 5.32 5.65
2400    4.01 3.88 3.53 3.53 3.37 3.28 3.11 3.42 3.77 3.78 3.83 3.90 3.78 3.83 3.80 3.90
2500    2.29 2.27 2.21 2.21 1.98 2.11 2.13 2.29 2.28 2.22 2.20 2.31 2.28 2.29 2.35 2.35
2600    0.77 0.77 0.77 0.77 0.77 0.77 0.77 0.77 0.77 0.77 0.77 0.77 0.77 0.77 0.77 0.77   1200-4200: Meteorological
2700       0.0047     0.0339     0.1034     0.5651     0.1725     0.0886     0.0317               data: output of STAR
2800       0.0036     0.0326     0.0970     0.5479     0.1462     0.1094     0.0634               code for Site B
2900       0.0055     0.0393     0.0895     0.4490     0.1396     0.1630     0.1141
3000       0.0044     0.0623     0.1159     0.4606     0.1431     0.1286     0.0850
3100       0.0098     0.0653     0.1244     0.4797     0.1099     0.1149     0.0960
3200       0.0085     0.0923     0.1288     0.4373     0.1169     0.1149     0.1013
3300       0.0062     0.0776     0.1080     0.4110     0.1188     0.1572     0.1214
3400       0.0079     0.0770     0.1080     0.4975     0.1085     0.1177     0.0833
3500       0.0131     0.0612     0.1093     0.5924     0.1028     0.0786     0.0428
3600       0.0086     0.0545     0.1053     0.6078     0.1248     0.0695     0.0295
3700       0.0044     0.0428     0.0853     0.6875     0.0881     0.0587     0.0332
```

Table A-4. Sample input for AIRDOS-II code--continued

```
3800      0.0050    0.0349    0.0840    0.6137    0.1217    0.0825    0.0583
3900      0.0069    0.0491    0.1115    0.4029    0.1563    0.1701    0.1032
4000      0.0057    0.0679    0.1657    0.4158    0.1589    0.1191    0.0669
4100      0.0091    0.0782    0.1609    0.4220    0.1729    0.1192    0.0377
4200      0.0057    0.0555    0.1287    0.5010    0.1776    0.1031    0.0284
4300 STACKS
4400          1
4500 10.0
4600 NUCLIDES
4700          2
4800 U-238
4900   4.3E-13    1.0E-5      .01
5000   1.0E-3
5100 RN-222
5200   1.81E-1
5300   1.0E+0
5400
5500
5600
5700
5800     0    1    2    5   17   34   47  135  189  539 1724 3367 47141347018858    0
5900     0    0    0    0    0    1    2    5   17   34   47  135  189  539 1724 3367
6000 47141347018858    0    0    0    0    0    0    1    2    5   17   34   47  135
6100   189  539 1724 3367 47141347018858    0    0    0    0    0    0    1    2    5
6200    17   34   47  135  189  539 1724 3367 47141347018858    0    0    0    0    0
6300     0    1    2    5   17   34   47  135  189  539 1724 3367 47141347018858    0
6400     0    0    0    0    0    1    2    5   17   34   47  135  189  539 1724 3367
6500 47141347018858    0    0    0    0    0    0    1    2    5   17   34   47  135
6600   189  539 1724 3367 47141347018858    0    0    0    0    0    0    1    2    5
6700    17   34   47  135  189  539 1724 3367 47141347018858    0    0    0    0    0
6800     0    1    2    5   17   34   47  135  189  539 1724 3367 47141347018858    0
6900     0    0    0    0    0    1    2    5   17   34   47  135  189  539 1724 3367
7000 47141347018858    0    0    0    0    0    0    1    2    5   17   34   47  135
7100   189  539 1724 3367 47141347018858    0    0    0    0    0    0    1    2    5
7200    17   34   47  135  189  539 1724 3367 47141347018858    0    0    0    0    0
7300     0    1    2    5   17   34   47  135  189  539 1724 3367 47141347018858    0
7400     0    0    0    0    0    1    2    5   17   34   47  135  189  539 1724 3367
```

4500: Physical stack height in meters

4700: Number of radionuclides in source term

4900: (1) Effective radiological decay constant in the plume (day^{-1}); (2) scavenging coefficient (sec^{-1}); (3) dry deposition velocity (m/sec)

5000: Release rate of radionuclide from stack (Ci/year)

5800–8200: Number of beef cattle for each environmental location (20 by 20 array). Read in sector 1 for each increasing distance from stack placing zeros at locations that are not considered in a given run of the AIRDOS-II code. Continue through sector 16 and place zeros in the remaining four sectors (circular option).

Table A-4. Sample input for AIRDOS-II code--continued

```
7500     47141347018858    0    0    0    0    0    0    1    2    5   17   34   47  135
7600      189  539 1724 3367 47141347018858    0    0    0    0    0    0    1    2    5
7700       17   34   47  135  189  539 1724 3367 47141347018858    0    0    0    0    0
7800        0    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0
7900        0    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0
8000        0    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0
8100        0    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0
8200        0    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0
8300        0    0    0    0    1    2    3    7   10   30   95  186  260  742 1039    0
8400        0    0    0    0    0    0    0    0    1    2    3    7   10   30   95  186
8500      260  742 1039    0    0    0    0    0    0    0    0    0    1    2    3    7
8600       10   30   95  186  260  742 1039    0    0    0    0    0    0    0    0    0
8700        1    2    3    7   10   30   95  186  260  742 1039    0    0    0    0    0
8800        0    0    0    0    1    2    3    7   10   30   95  186  260  742 1039    0
8900        0    0    0    0    0    0    0    0    1    2    3    7   10   30   95  186
9000      260  742 1039    0    0    0    0    0    0    0    0    0    1    2    3    7
9100       10   30   95  186  260  742 1039    0    0    0    0    0    0    0    0    0
9200        1    2    3    7   10   30   95  186  260  742 1039    0    0    0    0    0
9300        0    0    0    0    1    2    3    7   10   30   95  186  260  742 1039    0
9400        0    0    0    0    0    0    0    0    1    2    3    7   10   30   95  186
9500      260  742 1039    0    0    0    0    0    0    0    0    0    1    2    3    7
9600       10   30   95  186  260  742 1039    0    0    0    0    0    0    0    0    0
9700        1    2    3    7   10   30   95  186  260  742 1039    0    0    0    0    0
9800        0    0    0    0    1    2    3    7   10   30   95  186  260  742 1039    0
9900        0    0    0    0    0    0    0    0    1    2    3    7   10   30   95  186
10000     260  742 1039    0    0    0    0    0    0    0    0    0    1    2    3    7
10100      10   30   95  186  260  742 1039    0    0    0    0    0    0    0    0    0
10200       1    2    3    7   10   30   95  186  260  742 1039    0    0    0    0    0
10300       0    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0
10400       0    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0
10500       0    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0
10600       0    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0
10700       0    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0
```

8300–10700: Number of dairy cattle for each environmental location (20 by 20 array). Same input format as beef cattle.

Table A-4. Sample input for AIRDOS-II code--continued

```
10800  0.0       0.320E+03 0.448E+03 0.128E+04 0.409E+04 0.799E+04 0.112E+05 0.320E+05  10800-15700: Area of
10900  0.448E+05 0.128E+06 0.409E+06 0.799E+06 0.112E+07 0.320E+07 0.448E+07 0.0          vegecable crop
11000  0.0       0.0       0.0       0.0       0.0       0.320E+03 0.448E+03 0.128E+04    production for each
11100  0.409E+04 0.799E+04 0.112E+05 0.320E+05 0.448E+05 0.128E+06 0.409E+06 0.799E+06    environmental location
11200  0.112E+07 0.320E+07 0.448E+07 0.0       0.0       0.0       0.0       0.0          (20 by 20 array). Same
11300  0.0       0.320E+03 0.448E+03 0.128E+04 0.409E+04 0.799E+04 0.112E+05 0.320E+05    input format as beef
11400  0.448E+05 0.128E+06 0.409E+06 0.799E+06 0.112E+07 0.320E+07 0.448E+07 0.0          cattle
11500  0.0       0.0       0.0       0.0       0.0       0.320E+03 0.448E+03 0.128E+04
11600  0.409E+04 0.799E+04 0.112E+05 0.320E+05 0.448E+05 0.128E+06 0.409E+06 0.799E+06
11700  0.112E+07 0.320E+07 0.448E+07 0.0       0.0       0.0       0.0       0.0
11800  0.0       0.320E+03 0.448E+03 0.128E+04 0.409E+04 0.799E+04 0.112E+05 0.320E+05
11900  0.448E+05 0.128E+06 0.409E+06 0.799E+06 0.112E+07 0.320E+07 0.448E+07 0.0
12000  0.0       0.0       0.0       0.0       0.0       0.320E+03 0.448E+03 0.128E+04
12100  0.409E+04 0.799E+04 0.112E+05 0.320E+05 0.448E+05 0.128E+06 0.409E+06 0.799E+06
12200  0.112E+07 0.320E+07 0.448E+07 0.0       0.0       0.0       0.0       0.0
12300  0.0       0.320E+03 0.448E+03 0.128E+04 0.409E+04 0.799E+04 0.112E+05 0.320E+05
12400  0.448E+05 0.128E+06 0.409E+06 0.799E+06 0.112E+07 0.320E+07 0.448E+07 0.0
12500  0.0       0.0       0.0       0.0       0.0       0.320E+03 0.448E+03 0.128E+04
12600  0.409E+04 0.799E+04 0.112E+05 0.320E+05 0.448E+05 0.128E+06 0.409E+06 0.799E+06
12700  0.112E+07 0.320E+07 0.448E+07 0.0       0.0       0.0       0.0       0.0
12800  0.0       0.320E+03 0.448E+03 0.128E+04 0.409E+04 0.799E+04 0.112E+05 0.320E+05
12900  0.448E+05 0.128E+06 0.409E+06 0.799E+06 0.112E+07 0.320E+07 0.448E+07 0.0
13000  0.0       0.0       0.0       0.0       0.0       0.320E+03 0.448E+03 0.128E+04
13100  0.409E+04 0.799E+04 0.112E+05 0.320E+05 0.448E+05 0.128E+06 0.409E+06 0.799E+06
13200  0.112E+07 0.320E+07 0.448E+07 0.0       0.0       0.0       0.0       0.0
13300  0.0       0.320E+03 0.448E+03 0.128E+04 0.409E+04 0.799E+04 0.112E+05 0.320E+05
13400  0.448E+05 0.128E+06 0.409E+06 0.799E+06 0.112E+07 0.320E+07 0.448E+07 0.0
13500  0.0       0.0       0.0       0.0       0.0       0.320E+03 0.448E+03 0.128E+04
13600  0.409E+04 0.799E+04 0.112E+05 0.320E+05 0.448E+05 0.128E+06 0.409E+06 0.799E+06
13700  0.112E+07 0.320E+07 0.448E+07 0.0       0.0       0.0       0.0       0.0
13800  0.0       0.320E+03 0.448E+03 0.128E+04 0.409E+04 0.799E+04 0.112E+05 0.320E+05
13900  0.448E+05 0.128E+06 0.409E+06 0.799E+06 0.112E+07 0.320E+07 0.448E+07 0.0
14000  0.0       0.0       0.0       0.0       0.0       0.320E+03 0.448E+03 0.128E+04
14100  0.409E+04 0.799E+04 0.112E+05 0.320E+05 0.448E+05 0.128E+06 0.409E+06 0.799E+06
14200  0.112E+07 0.320E+07 0.448E+07 0.0       0.0       0.0       0.0       0.0
14300  0.0       0.320E+03 0.448E+03 0.128E+04 0.409E+04 0.799E+04 0.112E+05 0.320E+05
14400  0.448E+05 0.128E+06 0.409E+06 0.799E+06 0.112E+07 0.320E+07 0.448E+07 0.0
```

Table A-4. Sample input for AIRDOS-II code--continued

14500	0.0	0.0	0.0	0.0	0.0	0.320E+03	0.448E+03	0.128E+04	
14600	0.409E+04	0.799E+04	0.112E+05	0.320E+05	0.448E+05	0.128E+06	0.409E+06	0.799E+06	
14700	0.112E+07	0.320E+07	0.448E+07	0.0	0.0	0.0	0.0	0.0	
14800	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	
14900	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	
15000	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	
15100	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	
15200	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	
15300	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	
15400	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	
15500	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	
15600	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	
15700	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	
15800	0.	0.	0.	0.	0.	0.	0.	0.	15800-20700: Population
15900	1617.	0.	689.	8447.	432.	6598.	6353.	0.	for each environmental
16000	0.	0.	0.	0.	0.	0.	0.	0.	location (20 by 20
16100	0.	0.	0.	0.	0.	0.	776.	537.	array). Same input
16200	1314.	5118.	3149.	0.	0.	0.	0.	0.	format at beef cattle.
16300	0.	0.	0.	0.	0.	0.	0.	0.	
16400	142.	205.	805.	587.	519.	2809.	4225.	0.	
16500	0.	0.	0.	0.	0.	0.	0.	0.	
16600	0.	0.	0.	509.	0.	168.	1018.	340.	
16700	1374.	4197.	3226.	0.	0.	0.	0.	0.	
16800	0.	0.	0.	0.	0.	0.	0.	0.	
16900	382.	0.	1238.	1161.	1030.	10556.	9313.	0.	
17000	0.	0.	0.	0.	0.	0.	0.	0.	
17100	0.	0.	0.	0.	0.	0.	3558.	12185.	
17200	15267.	11705.	12121.	0.	0.	0.	0.	0.	
17300	0.	0.	0.	0.	0.	457.	0.	0.	
17400	0.	0.	1294.	51986.	15500.	8032.	14571.	0.	
17500	0.	0.	0.	0.	0.	0.	0.	0.	
17600	0.	0.	0.	0.	0.	12777.	65746.	140250.	
17700	109759.	75806.	16641.	0.	0.	0.	0.	0.	
17800	0.	0.	0.	0.	0.	0.	0.	0.	
17900	1584.	5796.	103746.	340629.	445726.	126650.	31141.	0.	
18000	0.	0.	0.	0.	0.	0.	0.	0.	
18100	0.	0.	0.	0.	0.	0.	48957.	105781.	

Table A-4. Sample input for AIRDOS-II code--continued

```
18200  152084.  106459.    9494.      0.       0.       0.       0.       0.
18300       0.       0.       0.      0.       0.       0.       0.       0.
18400       0.    1115.     232.  26816.   43201.   42605.   13021.       0.
18500       0.       0.       0.      0.       0.       0.       0.       0.
18600       0.       0.       0.      0.       0.    7612.   29628.    7761.
18700   14766.   13145.   13605.      0.       0.       0.       0.       0.
18800       0.       0.       0.      0.       0.       0.       0.       0.
18900    3830.   15804.   22332.   8238.    3409.    4183.    4931.       0.
19000       0.       0.       0.      0.       0.       0.       0.       0.
19100       0.       0.       0.    475.    1897.    7397.    1447.    3138.
19200    2823.   15826.   16398.      0.       0.       0.       0.       0.
19300       0.       0.       0.      0.       0.       0.     997.     799.
19400     564.    1251.    3269.   1843.    1135.    7474.    7535.       0.
19500       0.       0.       0.      0.       0.       0.       0.       0.
19600       0.       0.       0.    144.       0.       0.     634.     633.
19700    1245.    1874.    6481.      0.       0.       0.       0.       0.
19800       0.       0.       0.      0.       0.       0.       0.       0.
19900       0.       0.       0.      0.       0.       0.       0.       0.
20000       0.       0.       0.      0.       0.       0.       0.       0.
20100       0.       0.       0.      0.       0.       0.       0.       0.
20200       0.       0.       0.      0.       0.       0.       0.       0.
20300       0.       0.       0.      0.       0.       0.       0.       0.
20400       0.       0.       0.      0.       0.       0.       0.       0.
20500       0.       0.       0.      0.       0.       0.       0.       0.
20600       0.       0.       0.      0.       0.       0.       0.       0.
20700       0.       0.       0.      0.       0.       0.       0.       0.
20800
20900
21000
21100
21200
21300
21400
21500
21600
21700
```

Table A-4. Sample input for AIRDOS-II code--continued

```
21800          1
21900      0.076       0.0     0.008       0.0       0.0       0.0
22000 957950.000   152.400      .000    36525.
22100      180.
22200 1000.   10000.  0.1       0.15     20.0     0.25     1.0      1.0      1.0
22300 0.4      200.0  1.4       0.1      0.9      0.25     0.00381 2.0      0.3      1.0
22400 0.0495   0.0495  .0001096.0001096.0000000.00000005.5       0.25     10.0     11.0
22500          5  4.33E-13    1.199E0  2.534E-3  2.432E-3
22600  .150 .046 .146 .051 .045 .122 .033 .030 .031 .123 .028
22700     4.2E-5     2.5E-3      1.E0      5.E-4    6.93E-3     3.E-4     2.1E-5
22800
22900 TOT.BODY       2.1E-1  4.502E-2      CFINHA FROM DACRIN Y,1.0 AMAD
23000 BONE           3.6E+0  7.601E-1
23100 KIDNEYS        8.2E-1  1.733E-1
23200 LUNGS          3.9E+2  4.50E-2
23300 GI TRACT       3.0E-2  1.46E-1
23400          2  1.813E-1    3.95E-1   8.29E-8    7.02E-5
23500  .323 .592 1.00 .861 .766 .818 .728 .682 .864 .908 .301
23600   3.50E-13  3.50E-13   2.00E-2   1.89E-6  1.00E-11  7.00E-13
23700
23800 TOT.BODY      1.0E-60   1.0E-60
23900 WL             7.0E+0
```

21800-21900: Input parameters to characterize sources of vegetables, beef, and milk.

22100: Length of the grazing season

22200-22400: Ingestion pathway parameters

22500-23900: Dose conversion factor information

Table A-5. Site B meteorological data (STAR format)

```
                                                        STATION      PERIOD     SEQUENCE
(1)(2)  WS1    WS2    WS3    WS4    WS5    WS6          NUMBER (2) OF RECORD     NUMBER

  N A 0.000220.000320.000000.000000.000000.00000     13994  1A1760 16412         0603
NNE A 0.000140.000160.000000.000000.000000.00000     13994  2A1760 16412         0603
 NE A 0.000080.000160.000000.000000.000000.00000     13994  3A1760 16412         0603
ENE A 0.000120.000180.000000.000000.000000.00000     13994  4A1760 16412         0603
  E A 0.000160.000250.000000.000000.000000.00000     13994  5A1760 16412         0603
ESE A 0.000090.000180.000000.000000.000000.00000     13994  6A1760 16412         0603
 SE A 0.000180.000250.000000.000000.000000.00000     13994  7A1760 16412         0603
SSE A 0.000050.000250.000000.000000.000000.00000     13994  8A1760 16412         0603
  S A 0.000130.000390.000000.000000.000000.00000     13994  9A1760 16412         0603
SSW A 0.000070.000230.000000.000000.000000.00000     1399410A1760 16412         0603
 SW A 0.000150.000340.000000.000000.000000.00000     1399411A1760 16412         0603
WSW A 0.000100.000250.000000.000000.000000.00000     1399412A1760 16412         0603
  W A 0.000200.000340.000000.000000.000000.00000     1399413A1760 16412         0603
WNW A 0.000170.000340.000000.000000.000000.00000     1399414A1760 16412         0603
 NW A 0.000080.000250.000000.000000.000000.00000     1399415A1760 16412         0603
NNW A 0.000110.000300.000000.000000.000000.00000     1399416A1760 16412         0603
  N B 0.000540.001260.000730.000000.000000.00000     13994  1B1760 16412         0603
NNE B 0.000670.001370.000890.000000.000000.00000     13994  2B1760 16412         0603
 NE B 0.000590.001760.000660.000000.000000.00000     13994  3B1760 16412         0603
ENE B 0.000740.001740.000780.000000.000000.00000     13994  4B1760 16412         0603
  E B 0.000610.001260.000870.000000.000000.00000     13994  5B1760 16412         0603
ESE B 0.001140.001580.001100.000000.000000.00000     13994  6B1760 16412         0603
 SE B 0.000700.001550.000840.000000.000000.00000     13994  7B1760 16412         0603
SSE B 0.000330.001350.001070.000000.000000.00000     13994  8B1760 16412         0603
  S B 0.000800.001870.001070.000000.000000.00000     13994  9B1760 16412         0603
SSW B 0.000530.001510.000890.000000.000000.00000     1399410B1760 16412         0603
 SW B 0.000560.001960.001710.000000.000000.00000     1399411B1760 16412         0603
WSW B 0.000670.002190.001280.000000.000000.00000     1399412B1760 16412         0603
  W B 0.001150.001640.001070.000000.000000.00000     1399413B1760 16412         0603
WNW B 0.000610.001960.000960.000000.000000.00000     1399414B1760 16412         0603
 NW B 0.000740.001370.001100.000000.000000.00000     1399415B1760 16412         0603
NNW B 0.000610.001160.000820.000000.000000.00000     1399416B1760 16412         0603
  N C 0.000250.001300.002420.000550.000000.00000     13994  1C1760 16412         0603
NNE C 0.000190.001390.002210.000320.000000.00000     13994  2C1760 16412         0603
 NE C 0.000380.001370.001870.000550.000020.00000     13994  3C1760 16412         0603
ENE C 0.000390.001190.002650.000320.000000.00000     13994  4C1760 16412         0603
  E C 0.000240.001460.003220.000300.000000.00000     13994  5C1760 16412         0603
ESE C 0.000570.002210.003790.000530.000000.00000     13994  6C1760 16412         0603
 SE C 0.000530.002010.003860.000620.000020.00000     13994  7C1760 16412         0603
SSE C 0.000350.001620.005090.001030.000090.00000     13994  8C1760 16412         0603
  S C 0.000430.001990.007100.001670.000210.00000     13994  9C1760 16412         0603
SSW C 0.000170.000820.004860.000870.000070.00000     1399410C1760 16412         0603
 SW C 0.000350.002150.005000.001140.000070.00000     1399411C1760 16412         0603
WSW C 0.000420.002170.006160.001350.000000.00000     1399412C1760 16412         0603
  W C 0.000720.002240.004820.000910.000070.00000     1399413C1760 16412         0603
WNW C 0.000520.001920.004910.000980.000140.00002     1399414C1760 16412         0603
 NW C 0.000280.001370.003740.000960.000020.00002     1399415C1760 16412         0603
NNW C 0.000260.001120.003010.000570.000020.00002     1399416C1760 16412         0603
```

Table A-5. Site B meteorological data (STAR format)--continued

```
                                                          STATION     PERIOD    SEQUENCE
(1)(2)  WS1     WS2     WS3     WS4     WS5     WS6      NUMBER (2) OF RECORD    NUMBER

  N D 0.000640.003080.009160.010530.001070.00002         13994 1D1760 16412        0603
NNE D 0.000710.003470.007510.006710.000460.00007         13994 2D1760 16412        0603
 NE D 0.001060.003880.006830.003900.000230.00005         13994 3D1760 16412        0603
ENE D 0.000900.003970.006670.003700.000210.00000         13994 4D1760 16412        0603
  E D 0.000950.003720.009680.005270.000460.00005         13994 5D1760 16412        0603
ESE D 0.001020.004290.012560.009410.000870.00007         13994 6D1760 16412        0603
 SE D 0.000850.003630.014750.014230.001740.00011         13994 7D1760 16412        0603
SSE D 0.000580.003240.015210.023130.003720.00034         13994 8D1760 16412        0603
  S D 0.000830.003700.020890.032150.004340.00039         13994 9D1760 16412        0603
SSW D 0.000360.002330.010460.011620.001580.00009         1399410D1760 16412        0603
 SW D 0.000480.002900.009640.008810.000870.00014         1399411D1760 16412        0603
WSW D 0.000600.003220.008010.011670.001440.00041         1399412D1760 16412        0603
  W D 0.000830.004040.009360.012950.003290.00119         1399413D1760 16412        0603
WNW D 0.000880.003770.012060.030030.012860.00244         1399414D1760 16412        0603
 NW D 0.000780.004110.012810.026490.006210.00112         1399415D1760 16412        0603
NNW D 0.000460.002630.010640.012970.001940.00023         1399416D1760 16412        0603
  N E 0.000000.001440.002810.000000.000000.00000         13994 1E1760 16412        0603
NNE E 0.000000.002190.001940.000000.000000.00000         13994 2E1760 16412        0603
 NE E 0.000000.003240.001370.000000.000000.00000         13994 3E1760 16412        0603
ENE E 0.000000.002510.001620.000000.000000.00000         13994 4E1760 16412        0603
  E E 0.000000.002580.002030.000000.000000.00000         13994 5E1760 16412        0603
ESE E 0.000000.004110.004660.000000.000000.00000         13994 6E1760 16412        0603
 SE E 0.000000.005180.005800.000000.000000.00000         13994 7E1760 16412        0603
SSE E 0.000000.003400.008930.000000.000000.00000         13994 8E1760 16412        0603
  S E 0.000000.003880.015140.000000.000000.00000         13994 9E1760 16412        0603
SSW E 0.000000.002470.006900.000000.000000.00000         1399410E1760 16412        0603
 SW E 0.000000.003010.006350.000000.000000.00000         1399411E1760 16412        0603
WSW E 0.000000.002950.006740.000000.000000.00000         1399412E1760 16412        0603
  W E 0.000000.004060.008220.000000.000000.00000         1399413E1760 16412        0603
WNW E 0.000000.003240.009060.000000.000000.00000         1399414E1760 16412        0603
 NW E 0.000000.001990.004610.000000.000000.00000         1399415E1760 16412        0603
NNW E 0.000000.001960.003970.000000.000000.00000         1399416E1760 16412        0603
  N F 0.000530.002720.000000.000000.000000.00000         13994 1F1760 16412        0603
NNE F 0.000710.003770.000000.000000.000000.00000         13994 2F1760 16412        0603
 NE F 0.001490.004610.000000.000000.000000.00000         13994 3F1760 16412        0603
ENE F 0.001050.003010.000000.000000.000000.00000         13994 4F1760 16412        0603
  E F 0.001580.003240.000000.000000.000000.00000         13994 5F1760 16412        0603
ESE F 0.001600.006280.000000.000000.000000.00000         13994 6F1760 16412        0603
 SE F 0.002570.010250.000000.000000.000000.00000         13994 7F1760 16412        0603
SSE F 0.001560.007670.000000.000000.000000.00000         13994 8F1760 16412        0603
  S F 0.001550.008220.000000.000000.000000.00000         13994 9F1760 16412        0603
SSW F 0.000690.004750.000000.000000.000000.00000         1399410F1760 16412        0603
 SW F 0.000810.005640.000000.000000.000000.00000         1399411F1760 16412        0603
WSW F 0.001140.006120.000000.000000.000000.00000         1399412F1760 16412        0603
  W F 0.002180.011190.000000.000000.000000.00000         1399413F1760 16412        0603
WNW F 0.001240.007100.000000.000000.000000.00000         1399414F1760 16412        0603
 NW F 0.000910.003490.000000.000000.000000.00000         1399415F1760 16412        0603
NNW F 0.000650.002650.000000.000000.000000.00000         1399416F1760 16412        0603
```

Table A-5. Site B meteorological data (STAR format)--continued

```
                                                   STATION       PERIOD      SEQUENCE
(1)(2)  WS1     WS2     WS3     WS4     WS5     WS6          NUMBER (2) OF RECORD     NUMBER

  N G 0.001770.000000.000000.000000.000000.00000      13994  1G1760 16412         0603
NNE G 0.003170.000000.000000.000000.000000.00000      13994  2G1760 16412         0603
 NE G 0.004710.000000.000000.000000.000000.00000      13994  3G1760 16412         0603
ENE G 0.003580.000000.000000.000000.000000.00000      13994  4G1760 16412         0603
  E G 0.004030.000000.000000.000000.000000.00000      13994  5G1760 16412         0603
ESE G 0.005210.000000.000000.000000.000000.00000      13994  6G1760 16412         0603
 SE G 0.008970.000000.000000.000000.000000.00000      13994  7G1760 16412         0603
SSE G 0.005350.000000.000000.000000.000000.00000      13994  8G1760 16412         0603
  S G 0.003490.000000.000000.000000.000000.00000      13994  9G1760 16412         0603
SSW G 0.001500.000000.000000.000000.000000.00000      1399410G1760 16412         0603
 SW G 0.002040.000000.000000.000000.000000.00000      1399411G1760 16412         0603
WSW G 0.004080.000000.000000.000000.000000.00000      1399412G1760 16412         0603
  W G 0.008110.000000.000000.000000.000000.00000      1399413G1760 16412         0603
WNW G 0.005890.000000.000000.000000.000000.00000      1399414G1760 16412         0603
 NW G 0.002490.000000.000000.000000.000000.00000      1399415G1760 16412         0603
NNW G 0.001400.000000.000000.000000.000000.00000      1399416G1760 16412         0603
```

(1) - Wind direction (from indicated direction)
(2) - Stability class (A through G)
WS1 - Wind speed class 0 - 3 knots
WS2 - Wind speed class 4 - 6 knots
WS3 - Wind speed class 7 - 10 knots
WS4 - Wind speed class 11 - 16 knots
WS5 - Wind speed class 17 - 21 knots
WS6 - Wind speed class 21+ knots
Period of record: January 1960 - December 1964

Table A-6. Listing of STAR code

```
C     UTILITY PROGRAM TO CONVERT STAR FORMAT METEOROLOGICAL DATA TO
C     AIRDOS-II INPUT FORMAT.  MODIFIED BY C. B. NELSON  12/15/78.
C
      DIMENSION WDCS(16,8,6),WS(6),RWS(6),DICAT(16,8),TWDC(16,8),
     1RWDC(16,8),PERD(16)
      DATA WDCS/768*0./
      DO 1 IC=1,8
      DO 1 IS=1,16
      ID=MOD(25-IS,16)+1
    1 READ(50,200,END=2)(WDCS(ID,IC,IW),IW=1,6)
  200 FORMAT(T8,6F7.5)
    2 CONTINUE
      WS(1)=1.5*.5148
      WS(2)=5.*.5148
      WS(3)=8.5*.5148
      WS(4)=13.5*.5148
      WS(5)=19.*.5148
      WS(6)=23.*.5148
      DO 4 I=1,6
    4 RWS(I)=1./WS(I)
      DO 5 ID=1,16
      DO 6 IC=1,8
      DICAT(ID,IC)=0
      DO 7 IW=1,6
    7 DICAT(ID,IC)=WDCS(ID,IC,IW)+DICAT(ID,IC)
    6 CONTINUE
    5 CONTINUE
      DO 8 ID=1,16
      DO 9 IC=1,8
      SUM1=0
      SUM2=0
      SUM3=0
      DO 10 IW=1,6
      SUM1=SUM1+WDCS(ID,IC,IW)*WS(IW)
      SUM2=SUM2+WDCS(ID,IC,IW)*RWS(IW)
   10 SUM3=SUM3+WDCS(ID,IC,IW)
      IF(SUM3.EQ.0)TWDC(ID,IC)=0
      IF(SUM3.EQ.0)RWDC(ID,IC)=0
      IF(SUM3.EQ.0)GO TO 80
      TWDC(ID,IC)=SUM1/SUM3
      RWDC(ID,IC)=SUM2/SUM3
   80 CONTINUE
    9 CONTINUE
    8 CONTINUE
      DO 40 ID=1,16
      DO 41 IC=1,7
      IF(RWDC(ID,IC).EQ.0)GO TO 81
      RWDC(ID,IC)=1./RWDC(ID,IC)
   81 CONTINUE
   41 CONTINUE
   40 CONTINUE
```

Table A-6. Listing of STAR code--continued

```
      DO 50 ID=1,16
      PERD(ID)=0
      DO 51 IC=1,7
51    PERD(ID)=PERD(ID)+DICAT(ID,IC)
      DO 52 IC=1,7
52    DICAT(ID,IC)=DICAT(ID,IC)/PERD(ID)
50    CONTINUE
      SUM=0
      DO 70 ID=1,16
70    SUM=SUM+PERD(ID)
      DO 71 ID=1,16
71    PERD(ID)=PERD(ID)/SUM
      WRITE(51,300)(PERD(I),I=1,16)
300   FORMAT(' ',T20,16F5.3)
      WRITE(51,302)((RWDC(ID,IC),ID=1,16),IC=1,7)
302   FORMAT(' ',T20,16F5.2)
      WRITE(51,302)((TWDC(ID,IC),ID=1,16),IC=1,7)
      WRITE(51,301)((DICAT(ID,IC),IC=1,7),ID=1,16)
301   FORMAT(' ',T20,7F10.4)
      PUNCH 400,(PERD(I),I=1,16)
400   FORMAT(16F5.3)
      PUNCH 402,((RWDC(ID,IC),ID=1,16),IC=1,7)
402   FORMAT(16F5.2)
      PUNCH 402,((TWDC(ID,IC),ID=1,16),IC=1,7)
      PUNCH 401,((DICAT(ID,IC),IC=1,7),ID=1,16)
401   FORMAT(7F10.4)
      STOP
      END
```

Table A-7. Cattle densities and vegetable crop distributions for use with AIRDOS-II

State	Dairy cattle density #/km^2	Beef cattle density #/km^2	Vegetable crop fraction km^2/km^2
Alabama	7.02E-1	1.52E+1	4.16E-3
Arizona	2.80E-1	3.73	2.90E-3
Arkansas	5.90E-1	1.27E+1	1.46E-3
California	2.85	8.81	1.18E-2
Colorado	3.50E-1	1.13E+1	1.39E-2
Connecticut	2.50E-1	3.60	7.93E-3
Delaware	2.72	6.48	5.85E-2
Florida	1.37	1.28E+1	6.92E-3
Georgia	8.63E-1	1.43E+1	2.17E-3
Idaho	8.56E-1	7.19	7.15E-2
Illinois	2.16	3.33E+1	2.80E-2
Indiana	2.80	3.34E+1	2.72E-2
Iowa	3.14	7.40E+1	2.43E-2
Kansas	8.00E-1	2.90E+1	5.97E-2
Kentucky	2.57	2.65E+1	3.98E-3
Louisiana	9.62E-1	1.08E+1	4.35E-2
Maine	8.07E-1	7.65E-1	5.97E-2
Maryland	6.11	1.09E+1	1.11E-2
Massachusetts	3.13	2.90	4.96E-3
Michigan	3.51	7.90	1.70E-2
Minnesota	4.88	1.85E+2	3.05E-2
Mississippi	8.70E-1	1.75E+1	1.07E-3
Missouri	1.89	3.43E+1	8.14E-3
Montana	9.27E-2	7.29	8.78E-3
Nebraska	8.78E-1	3.50E+1	2.39E-2
Nevada	5.65E-2	1.84	8.92E-3
New Hampshire	1.58	1.40	6.69E-2
New Jersey	3.29	4.25	1.82E-2
New Mexico	1.14E-1	4.13	1.38E-3
New York	8.56	5.83	1.88E-2

Table A-7. Cattle densities and vegetable crop distributions for use with AIRDOS-II--continued

State	Dairy cattle density #/km^2	Beef cattle density #/km^2	Vegetable crop fraction km^2/km^2
North Carolina	1.26	1.02E+1	6.32E-3
North Dakota	6.25E-1	1.18E+1	6.29E-2
Ohio	4.56	2.03E+1	1.70E-2
Oklahoma	7.13E-1	2.68E+1	2.80E-2
Oregon	4.53E-1	4.56	1.59E-2
Pennsylvania	6.46	9.63	1.32E-2
Rhode Island	2.30	2.50	4.54E-2
South Carolina	7.02E-1	8.87	1.84E-3
South Dakota	8.85E-1	2.32E+1	1.20E-2
Tennessee	2.00E-1	2.11E+1	2.72E-3
Texas	5.30E-1	1.90E+1	5.77E-3
Utah	4.46E-1	2.84	1.83E-3
Vermont	8.88	4.71	1.08E-3
Virginia	1.84	1.31E+1	8.70E-3
Washington	1.50	5.62	5.20E-2
West Virgina	6.00E-1	6.23	1.16E-3
Wisconsin	1.43E+1	1.81E+1	1.78E-2
Wyoming	5.79E-2	5.12	1.59E-3

Table A-8. Site independent parameters used for AIRDOS-II generic site assessments

Symbolic variable	Description	Value
GRAZ	Length of grazing season	180. days
PTPMV	Period of time from vegetable production to human consumption	0.0 days
PTPMB	Period of time from beef production to human consumption	0.0 days
PTPMM	Period of time from milk production to human consumption	0.0 days
BRTHRT	Human breathing rate	9.58E+5 cm^3/h
T	Buildup time for surface deposition	3.65E+4 days
A	Soil surface area to furnish food crops for one person	1.0E+3 m^2
ASUBG	Pasture area per cow	1.0E+4 m^2
DSUBF	Dry weight area density of above surface food	1.0E-1 kg/m^2
DSUBG	Dry weight area grass density	1.5E-1 kg/m^2
SMALLD	Depth of plow layer	2.0E+1 cm
KSUBB	Rate of increase of steer muscle mass	4.0E-1 kg/day
MSUBB	Muscle mass of steer at slaughter	2.0E+2 kg
RHO	Soil density	1.4 gm/cm^3
S1	Fraction of deposited radionuclides intercepted by above-surface food crop	1.0E-1

Table A-8. Site independent parameters used for AIRDOS-II generic site assessments--continued

Symbolic variable	Description	Value
S2	Fraction of deposited radionuclides on soil surface below above-surface food crop	9.0E-1
S3	Fraction of deposited radionuclides intercepted by pasture grass	2.5E-1
TAUBEF	Fraction of beef herd slaughtered per day	3.81E-3 day^{-1}
TAUMLK	Number of milkings per cow	2.0 day^{-1}
TAUBM	Human beef consumption rate	3.0E-1 kg/day
TAUCM	Human milk consumption rate	1.0 L/day
TAUES	Above-surface food crop to soil surface transfer rate	4.95E-2 day^{-1}
TAUGR	Pasture grass to pasture soil transfer rate	4.95E-2 day^{-1}
TAUPD	Soil pool to soil sink transfer rate	1.1E-4 day^{-1}
TAURD	Pasture soil to soil sink transfer rate	1.1E-4 day^{-1}
TAURG	Pasture soil to pasture grass transfer rate	0.0 day^{-1}
TAUSP	Soil surface to soil pool transfer rate	0.0 day^{-1}
U	Milk capacity of udder	5.5 L
V	Human consumption rate of vegetable food	2.5E-1 kg/day

Table A-8. Site independent parameters used for AIRDOS-II generic site assessments--continued

Symbolic variable	Description	Value
VSUBC	Grass consumption rate (dry) of cow	10. kg/day
VSUBM	Milk production rate of cow	11. L/day
DD1	Dietary correction factor for above-surface food	1.0
DD2	Dietary correction factor for uptake from soil	1.0
DD3	Dietary correction factor for beef	1.0
DD4	Dietary correction factor for milk	1.0

Table A-9. Dose conversion factors for use in preliminary Clean Air Act assessments

Radionuclide	Total body	Lung	Bone	Kidney	Thyroid	Liver	G. I. tract	Testis	Ovary	Spleen	Muscle
				Inhalation--rem per microcurie[a]							
Americium-241	1.4E+2	4.7E+2	8.3E+2			3.6E+2					
Plutonium-242	1.5E+2	1.7E+2	6.1E+3	6.5E+2		8.6E+2	4.3E-2				
Plutonium-241	2.5	1.6E-1	1.2E+2	1.2E+1		6.4	1.1E-3				
Plutonium-240	1.6E+2	1.8E+2	6.6E+3	6.8E+2		9.0E+2	4.3E-2				
Plutonium-239	1.6E+2	4.5E+2	7.7E+2	6.8E+2		3.5E+2	4.3E-2				
Plutonium-238	1.4E+2	1.9E+2	5.7E+3	6.1E+2		8.2E+2	4.3E-2				
Uranium-238	2.1E-1	3.9E+2	3.6	8.2E-1			3.0E-2				
Uranium-236	3.0E-1	2.6E+2	1.9E+1	8.2E-1			5.2E-2				
Uranium-235	2.3E-1	4.2E+2	3.8	8.8E-1			3.7E-2				
Uranium-234	2.4E-1	4.5E+2	3.9	9.4E-1			3.4E-2				
Uranium-233	1.3	5.5E+1	2.2E+1	5.1E+1			3.6E-2				
Protactinium-234m	1.7E-7	1.2E-5	7.0E-7	1.6E-6		3.5E-7	6.1E-5				
Protactinium-234	8.1E-5	1.5E-3	3.9E-4	3.4E-4		9.3E-5					
Thorium-234	9.9E-5	2.2E-1	3.4E-3	1.1E-3		2.0E-4	5.1E-2				
Thorium-232	3.2E+1	5.9E+2	9.4E+2	2.1E+2		4.4E+1	2.8E-2				
Thorium-231	7.5E-7	1.6E-3	1.6E-5	6.8E-6		1.4E-6	2.7E-3				
Thorium-230	2.5E+1	4.4E+2	8.3E+2	2.5E+2		5.2E+1	3.3E-2				
Thorium-228	3.0	1.8E+3	8.9E+1	8.5		1.5	3.5E-2				
Radium-228	8.6E+1	1.1E+3	8.1E+1				2.2E-4				
Radium-226	2.0	1.0E+3	2.0E+2				2.4E-2				
Radium-224	1.3E-1	2.1E+1	6.7E-1				6.2E-2				
Bismuth-214	0.0										
Bismuth-212	1.2E-5	1.0E-1	1.9E-5	8.6E-4		7.7E-5	1.3E-6			5.8E-5	
Lead-214	0.0										
Lead-212	8.4E-4	9.7E-1	2.9E-4	2.7E-2		2.8E-3	2.6E-3				
Lead-210	5.6E-1	3.7E+2	1.7E+1	1.4E+1		4.5	1.2E-3				
Polonium-210	5.1E-2	1.1E+2	2.2E-1	1.6		4.7E-1	3.0E-2				
Thallium-208	0.0										

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[a] Fifty-year dose commitment from one year's intake.

Table A-9. Dose conversion factors for use in preliminary Clean Air Act assessments--continued

Radionuclide	Total body	Lung	Bone	Kidney	Thyroid	Liver	G. I. tract	Testis	Ovary	Spleen	Muscle
					Inhalation--rem per microcurie[a] (continued)						
Xenon-138	1.0E-11	7.9E-6									
Xenon-135m	1.0E-11	2.3E-6									
Xenon-135	1.0E-11	4.0E-6									
Xenon-133m	1.0E-11	1.8E-6									
Xenon-133	1.0E-12	1.4E-6									
Xenon-131m	1.0E-11	1.4E-6									
Cesium-137	3.2E-2	9.5E-3	6.1E-2	2.8E-2		7.9E-2	1.1E-3			6.7E-2	6.0E-2
Cesium-134	5.6E-2	1.2E-2	3.7E-2	3.4E-2		1.0E-1	2.1E-3			7.5E-2	9.7E-2
Iodine-133	5.7E-4	5.1E-3			3.9E-1		2.1E-3				
Iodine-131	3.4E-4				8.3E-1						
Iodine-129	7.0E-3	9.0E-2			5.7		5.3E-3				
Iodine-125	2.6E-4				5.8E-1						
Technetium-99m	1.1E-5	1.8E-4	9.1E-6	8.4E-6	5.7E-5	1.5E-5	1.4E-5	2.6E-6	7.0E-6		
Technetium-99	8.9E-4	5.2E-2	2.4E-4	3.1E-4	9.5E-3	4.2E-4	1.7E-3	2.1E-4	2.1E-4		
Strontium-90	2.4E-1	9.9E-3	3.0	1.5E-2	1.5E-2	1.5E-2	1.4E-2	1.5E-2	1.5E-2		
Krypton-88	1.0E-11	8.0E-6									
Krypton-87	1.0E-11	1.6E-5									
Krypton-85m	1.0E-11	2.9E-6									
Krypton-85	1.0E-12	2.3E-6									
Krypton-83m	1.0E-11	4.4E-7									
Cobalt-60	8.2E-2	1.3	5.1E-2	5.9E-2	6.0E-2	1.2E-1	2.9E-2	6.1E-3	1.9E-2	1.5E-3	
Cobalt-58	4.2E-3	5.9E-2	2.6E-3	2.8E-3	3.2E-3	6.0E-3	7.4E-3	3.6E-4	2.4E-3	5.1E-4	
Iron-55	9.5E-4	6.4E-3	1.4E-3			4.1E-3	5.3E-4			7.4E-3	
Argon-41	1.0E-11	8.0E-6									
Carbon-14	1.4E-5	6.2E-6	8.5E-7	7.9E-6	6.5E-6	8.9E-6	7.2E-6	5.4E-6	5.3E-6		
Tritium	1.1E-4										

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[a] Fifty-year dose commitment from one year's intake.

Table A-9. Dose conversion factors for use in preliminary Clean Air Act assessments--continued

Radionuclide	Total body	Lung	Bone	Kidney	Thyroid	Liver	G. I. tract	Testis	Ovary	Spleen	Muscle
					Ingestion--rem per microcurie[a]						
Americium-241	4.7E-2		1.9E+1			8.3					
Plutonium-242	1.8E-2	1.8E-2	7.3E-1	7.8E-2		1.0E-1	6.5E-2				
Plutonium-241	3.1E-4	3.1E-4	1.5E-2	1.5E-3		7.6E-4	1.9E-3				
Plutonium-240	1.9E-2	1.9E-2	7.9E-1	8.1E-2		1.1E-1	6.5E-2				
Plutonium-239	4.4E-3	1.9E-2	1.9	8.1E-2		8.1E-1	6.5E-2				
Plutonium-238	1.7E-2	1.7E-2	6.8E-1	7.3E-2		9.7E-2	6.5E-2				
Uranium-238	4.5E-2	4.5E-2	7.6E-1	1.7E-1			1.5E-1				
Uranium-236	6.6E-2	3.1E-5	7.5E-1	3.2E-2			8.7E-2				
Uranium-235	4.8E-2	4.8E-2	7.9E-1	1.9E-1			6.5E-2				
Uranium-234	5.1E-2	5.1E-2	8.3E-1	2.0E-1			6.5E-2				
Uranium-233	5.2E-2	5.2E-2	8.6E-1	2.0E-1			6.5E-2				
Protactinium-234m	6.9E-11	6.9E-11	2.8E-10	6.2E-10		1.3E-10	9.7E-5				
Protactinium-234	3.2E-8	3.2E-8	1.6E-7	1.3E-7		3.6E-8					
Thorium-234	2.3E-6		7.9E-5	2.6E-5		4.7E-6	9.7E-2				
Thorium-232	7.4E-2	7.4E-2	2.3	4.8		1.0E-1	4.9E-1				
Thorium-231	2.0E-8	2.0E-8	4.4E-7	1.8E-7		3.7E-8	9.7E-3				
Thorium-230	5.7E-2		2.0	5.6E-1		1.2E-1	6.5E-2				
Thorium-228	1.7E-2	1.7E-2	4.9E-1	4.6E-2		8.3E-3	1.9E-1				
Radium-228	2.5	2.5	1.9E+1				6.5E-2				
Radium-226	1.0E-1		1.0E+1	3.0E-1		3.0E-1	6.5E-2				
Radium-224	3.1E-2	3.1E-2	2.4E-1				3.9E-1				
Bismuth-214	0.0										
Bismuth-212	3.7E-5	3.7E-5	5.4E-5	2.5E-3		2.3E-4	4.9E-3			1.7E-4	
Lead-214	0.0										
Lead-212	3.0E-3	3.0E-3	3.8E-2	8.7E-2		1.0E-2	9.7E-3				
Lead-210	3.3	5.2E-1	4.5E+1	4.6E-1		6.4E-1	9.7E-3				
Polonium-210	2.9E-1	8.6E-2	1.3	4.8		8.2E-1	6.5E-2			2.3	
Thallium-208	0.0										

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[a] Fifty-year dose commitment from one year's intake.

Table A-9. Dose conversion factors for use in preliminary Clean Air Act assessments--continued

Radionuclide	Total body	Lung	Bone	Kidney	Thyroid	Liver	G. I. tract	Testis	Ovary	Spleen	Muscle
Ingestion--rem per microcurie[a] (continued)											
Xenon-138	0.0										
Xenon-135m	0.0										
Xenon-135	0.0										
Xenon-133m	0.0										
Xenon-133	0.0										
Xenon-131m	0.0										
Cesium-137	4.3E-2	1.2E-2	8.1E-2	3.8E-2		1.1E-1	4.9E-2			8.8E-2	8.0E-2
Cesium-134	7.5E-2	1.5E-2	5.0E-2	4.5E-2		1.4E-1	4.9E-2			9.9E-2	1.3E-1
Iodine-133	7.7E-4	7.7E-4			5.1E-1		3.2E-3				
Iodine-131	4.5E-4				1.1						
Iodine-129	9.3E-3	9.3E-3			7.5		9.7E-3				
Iodine-125	3.4E-4				7.7E-1						
Technetium-99m	1.5E-5	8.5E-6	1.6E-5	2.5E-5	2.2E-4	2.8E-5	8.8E-5	1.1E-5	3.8E-5		
Technetium-99	2.1E-4		3.6E-4	4.6E-4	1.4E-2	6.3E-4	3.2E-3	3.2E-4	3.2E-4		
Strontium-90	9.5E-2	5.9E-9	1.2	6.0E-3	6.0E-3	5.7E-3	7.8E-2	6.0E-3	6.0E-3		
Krypton-88	0.0										
Krypton-87	0.0										
Krypton-85m	0.0										
Krypton-85	0.0										
Krypton-83m	0.0										
Cobalt-60	4.5E-3	4.5E-3		7.8E-4		2.0E-3	6.5E-2			1.1E-3	
Cobalt-58	1.6E-3	1.6E-3		2.7E-4		7.4E-4	2.2E-2			3.8E-4	
Iron-55	3.2E-4	7.8E-4	4.5E-4			1.3E-3	9.7E-4			2.5E-3	
Argon-41	0.0	0.0									
Carbon-14	1.9E-3	8.5E-4	1.2E-3	1.1E-3	8.9E-4	1.2E-3	1.5E-3	7.2E-4	7.4E-4		
Tritium	6.2E-5										

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[a] Fifty-year dose commitment from one year's intake.

Table A-9. Dose conversion factors for use in preliminary Clean Air Act assessments--continued

Radionuclide	Total body	Lung	Bone	Kidney	Thyroid	Liver	G. I. tract	Testis	Ovary	Spleen	Muscle
Air Submersion--rem-cm^3/microcurie-h											
Americium-241	1.6E+1	1.2E+1	3.1E+1	1.0E+1	1.6E+1	9.5	6.5	1.4E+1	9.7	9.2	1.4E+1
Plutonium-242	1.6E-1	2.7E-2	1.2E-1	1.5E-2	2.1E-2	1.0E-2	4.1E-2	1.3E-1	1.7E-2	1.5E-2	1.3E-1
Plutonium-241	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Plutonium-240	1.7E-1	3.3E-2	1.3E-1	2.0E-2	2.9E-2	1.6E-2	4.6E-2	1.3E-1	2.1E-2	2.0E-2	1.4E-1
Plutonium-239	8.2E-2	1.9E-2	7.3E-2	1.4E-2	2.1E-2	1.1E-2	2.1E-2	6.9E-2	1.3E-2	1.2E-2	6.7E-2
Plutonium-238	1.9E-1	3.4E-2	1.4E-1	2.0E-2	2.7E-2	1.5E-2	4.9E-2	1.5E-1	2.1E-2	2.1E-2	1.5E-1
Uranium-238	1.8E-1	5.4E-2	1.8E-1	4.0E-2	6.1E-2	3.6E-2	5.5E-2	1.5E-1	3.4E-2	3.7E-2	1.5E-1
Uranium-236	1.9E-1	4.0E-2	1.5E-1	2.6E-1	3.8E-2	2.1E-2	5.0E-2	1.4E-1	2.8E-2	2.5E-2	1.4E-1
Uranium-235	1.3E+2	1.1E+2	2.0E+2	9.2E+1	1.6E+2	9.6E+1	6.9E+1	1.6E+2	6.5E+1	9.7E+1	1.2E+2
Uranium-234	2.9E-1	1.2E-1	3.5E-1	9.6E-2	1.5E-1	9.0E-2	1.0E-1	2.5E-1	7.8E-2	8.6E-2	2.4E-1
Uranium-233	7.6E-1	5.7E-1	1.1	4.7E-1	6.7E-1	4.9E-1	3.7E-1	8.6E-1	3.5E-1	5.0E-1	6.7E-1
Protactinium-234m	7.6	7.0	8.6	7.0	5.9	6.3	4.8	6.9	4.5	7.4	7.5
Protactinium-234	2.6E+3	2.4E+3	2.8E+3	2.1E+3	2.3E+3	2.2E+3	2.0E+3	1.8E+3	2.3E+3	2.0E+3	2.5E+3
Thorium-234	6.0	4.9	1.1E+1	4.2	7.1	4.3	3.0	5.5	2.6	3.7	5.2
Thorium-232	2.1E-1	7.8E-2	2.4E-1	5.8E-2	8.8E-2	5.4E-2	6.7E-2	1.6E-1	6.1E-2	5.9E-2	1.7E-1
Thorium-231	1.4E+1	1.0E+1	2.3E+1	8.9	1.5E+1	8.7	6.2	1.4E+1	5.4	7.2	1.2E+1
Thorium-230	4.2E-1	2.4E-1	6.2E-1	2.0E-1	3.0E-1	1.9E-1	1.7E-1	3.7E-1	1.8E-1	1.9E-1	3.5E-1
Thorium-228	2.1	1.7	3.4	1.4	2.2	1.5	1.1	2.2	9.3E-1	1.3	1.8
Radium-228	1.4E+1	7.2	2.2E+1	8.0	1.5E+1	5.3	1.7	1.9E+1	5.1	4.8	1.3E+1
Radium-226	5.6	4.9	8.5	4.0	5.4	4.3	3.1	7.1	3.0	4.5	5.1
Radium-224	8.2	7.3	1.3E+1	6.0	8.0	6.3	4.6	1.1E+1	4.5	6.7	7.5
Bismuth-214	1.4E+3	1.3E+3	1.5E+3	1.1E+3	1.3E+3	1.2E+3	1.1E+3	1.1E+3	1.1E+3	1.4E+3	1.4E+3
Bismuth-212	9.2E+1	8.6E+1	1.1E+2	7.6E+1	9.0E+1	7.6E+1	6.9E-1	8.8E+1	5.5E+1	9.8E+1	9.1E+1
Lead-214	1.9E+2	1.8E+2	2.7E+2	1.6E+2	2.0E+2	1.6E+2	1.2E+2	2.3E+2	8.7E+1	1.8E+2	1.9E+2
Lead-212	1.2E+2	1.1E+2	1.9E+2	8.7E+1	1.2E+2	9.0E+1	6.5E+1	1.5E+2	6.6E+1	9.4E+1	1.1E+2
Lead-210	1.6	1.0	2.8	8.8E-1	1.4	8.4E-1	6.1E-1	1.4	8.7E-1	8.4E-1	1.4
Polonium-210	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Thallium-208	2.8E+3	2.7E+3	3.0E+3	2.2E+3	2.6E+3	2.5E+3	2.3E+3	1.8E+3	2.6E+3	2.0E+3	2.8E+3

Table A-9. Dose conversion factors for use in preliminary
Clean Air Act assessments--continued

Radionuclide	Total body	Lung	Bone	Kidney	Thyroid	Liver	G. I. tract	Testis	Ovary	Spleen	Muscle
Air Submersion--rem-cm^3/microcurie-h (continued)											
Xenon-138	6.8E+2	6.4E+2	7.8E+2	5.2E+2	6.6E+2	5.8E+2	5.4E+2	5.4E+2	5.7E+2	6.0E+2	6.7E+2
Xenon-135m	3.5E+2	3.3E+2	4.3E+2	3.1E+2	3.7E+2	2.9E+2	2.5E+2	3.9E+2	1.3E+2	3.7E+2	3.5E+2
Xenon-135	2.0E+2	1.8E+2	3.1E+2	1.5E+2	2.0E+2	1.6E+2	1.2E+2	2.6E+2	1.1E+2	1.7E+2	1.9E+2
Xenon-133m	2.7E+1	2.1E+1	4.1E+1	1.8E+1	2.6E+1	1.8E+1	1.2E+1	3.6E+1	1.3E+1	1.8E+1	2.5E+1
Xenon-133	3.3E+1	2.5E+1	5.6E+1	2.3E+1	3.9E+1	2.2E+1	1.4E+1	3.3E+1	1.3E+1	1.8E+1	2.9E+1
Xenon-131m	9.9	6.0	1.5E+1	6.0	1.0E+1	4.8	2.3	1.3E+1	4.0	4.6	9.2
Cesium-137	4.7E+2	4.4E+2	5.7E+2	4.1E+2	4.9E+2	3.9E+2	3.4E+2	5.2E+2	1.7E+2	4.9E+2	4.7E+2
Cesium-134	1.3E+3	1.2E+3	1.5E+3	1.2E+3	1.1E+3	1.1E+3	8.8E+2	1.3E+3	6.5E+2	1.3E+3	1.3E+3
Iodine-133	5.6E+2	5.3E+2	6.6E+2	4.9E+2	5.6E+2	4.7E+2	4.1E+2	5.8E+2	2.6E+2	5.9E+2	5.6E+2
Iodine-131	3.1E+2	2.9E+2	3.8E+2	2.7E+2	3.3E+2	2.6E+2	2.2E+2	3.5E+2	1.2E+2	3.2E+2	3.1E+2
Iodine-129	1.4E+1	6.8	2.0E+1	7.6	1.4E+1	5.1	1.6	1.8E+1	4.8	4.6	1.2E+1
Iodine-125	1.8E+1	9.2	2.8E+1	1.0E+1	1.9E+1	6.8	2.2	2.5E+1	6.5	6.2	1.7E+1
Technetium-99m	1.1E+2	9.1E+1	2.0E+2	7.9E+1	1.3E+2	8.1E+1	5.7E+1	1.0E+1	4.6E+1	6.6E+1	9.5E+1
Technetium-99	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Strontium-90	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Krypton-88	1.7E+3	1.6E+3	1.8E+3	1.3E+3	1.6E+3	1.5E+3	1.4E+3	1.0E+3	1.7E+3	1.2E+3	1.7E+3
Krypton-87	6.5E+2	6.2E+2	7.1E+2	5.3E+2	6.3E+2	5.7E+2	5.2E+2	4.7E+2	5.3E+2	5.4E+2	6.5E+2
Krypton-85m	1.3E+2	1.2E+2	2.0E+2	9.5E+1	1.3E+2	1.0E+2	7.3E+1	1.7E+2	7.1E+1	1.1E+2	1.2E+2
Krypton-85	1.7	1.6	2.1	1.5	1.8	1.4	1.2	1.9	6.2E-1	1.4	6.2E-1
Krypton-83m	2.7E-1	3.6E-2	1.6E-1	1.1E-2	1.6E-2	8.2E-3	6.3E-2	2.0E-1	1.9E-2	2.5E-2	2.0E-1
Cobalt-60	2.1E+3	2.0E+3	2.3E+3	1.7E+3	1.9E+3	1.8E+3	1.6E+3	1.7E+3	1.7E+3	2.3E+3	2.1E+3
Cobalt-58	7.7E+2	7.3E+2	8.6E+2	7.2E+2	6.2E+2	6.5E+2	5.1E+2	7.2E+2	4.5E+2	7.8E+2	7.7E+2
Iron-55	8.6E-2	6.4E-3	1.3E-2	1.6E-3	3.2E-3	1.6E-3	3.2E-3	6.2E-2	3.2E-3	1.8E-2	2.9E-2
Argon-41	1.1E+3	1.1E+3	1.1E+3	1.1E+3	1.1E+3	1.1E+3	1.1E+3	1.1E+3	1.1E+3	1.1E+3	1.1E+3
Carbon-14	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Tritium	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0

Table A-9. Dose conversion factors for use in preliminary Clean Air Act assessments--continued

Radionuclide	Total body	Lung	Bone	Kidney	Thyroid	Liver	G. I. tract	Testis	Ovary	Spleen	Muscle
Surface Exposure--rem-cm^2/microcurie-h											
Americium-241	8.6E-3	6.0E-3	1.6E-2	5.2E-3	8.2E-3	4.9E-3	3.4E-3	7.5E-3	5.1E-3	4.8E-3	7.3E-3
Plutonium-242	3.1E-4	5.3E-5	2.3E-4	3.0E-5	4.1E-5	2.0E-5	8.1E-5	2.5E-4	3.3E-5	3.0E-5	2.5E-4
Plutonium-241	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Plutonium-240	3.2E-4	6.4E-5	2.6E-4	3.8E-5	5.6E-5	3.1E-5	8.9E-5	2.6E-4	4.1E-5	3.8E-5	2.6E-4
Plutonium-239	1.5E-4	3.6E-5	1.4E-4	2.6E-5	4.0E-5	2.0E-5	4.0E-5	1.3E-4	2.4E-5	2.3E-5	1.2E-4
Plutonium-238	3.8E-4	6.9E-5	2.8E-4	3.9E-5	5.4E-5	3.0E-5	9.9E-5	3.0E-4	4.2E-5	4.2E-5	3.0E-4
Uranium-238	3.6E-4	1.1E-4	3.5E-4	7.9E-5	1.2E-4	7.2E-5	1.1E-4	3.0E-4	6.7E-5	7.4E-5	2.9E-4
Uranium-236	3.8E-4	8.2E-5	3.2E-4	5.3E-4	7.7E-5	4.4E-5	1.0E-4	2.9E-4	5.8E-5	5.2E-5	3.0E-4
Uranium-235	5.7E-2	5.0E-2	8.9E-2	4.1E-2	7.2E-2	4.3E-2	3.1E-2	7.0E-2	2.9E-2	4.4E-2	5.2E-2
Uranium-234	5.5E-4	2.3E-4	6.5E-4	1.8E-4	2.8E-4	1.7E-4	1.9E-4	4.6E-4	1.5E-4	1.6E-4	4.5E-4
Uranium-233	8.9E-4	6.8E-4	1.3E-3	5.6E-4	7.8E-4	5.7E-4	4.4E-4	1.0E-3	4.1E-4	5.8E-4	7.9E-4
Protactinium-234m	1.4E-3	1.3E-3	1.6E-3	1.3E-3	1.1E-3	1.2E-3	9.1E-4	1.3E-3	8.5E-4	1.4E-3	1.4E-3
Protactinium-234	4.2E-1	4.0E-1	4.6E-1	3.4E-1	3.8E-1	3.7E-1	3.3E-1	2.9E-1	3.7E-1	3.2E-1	4.2E-1
Thorium-234	1.9E-3	1.6E-3	3.4E-3	1.3E-3	2.3E-3	1.4E-3	9.6E-4	1.8E-3	8.3E-4	1.2E-3	1.6E-3
Thorium-232	4.1E-4	1.6E-4	4.8E-4	1.2E-4	1.8E-4	1.1E-4	1.3E-4	3.1E-4	1.2E-4	1.2E-4	3.3E-4
Thorium-231	1.5E-2	1.1E-2	2.4E-2	9.3E-3	1.6E-2	9.1E-3	6.6E-3	1.5E-2	5.7E-3	7.6E-3	1.3E-2
Thorium-230	6.9E-4	4.1E-4	1.0E-3	3.3E-4	5.1E-4	3.2E-4	2.8E-4	6.2E-4	3.0E-4	3.2E-4	5.8E-4
Thorium-228	1.9E-3	1.5E-3	3.0E-3	1.3E-3	2.0E-3	1.3E-3	9.6E-4	1.9E-3	8.2E-4	1.2E-3	1.6E-3
Radium-228	7.5E-3	3.8E-3	1.1E-2	4.3E-3	7.9E-3	2.8E-3	9.0E-4	1.0E-2	2.7E-3	2.6E-3	7.0E-3
Radium-226	1.1E-3	9.4E-4	1.6E-3	7.7E-4	1.0E-3	8.1E-4	5.9E-4	1.4E-3	5.7E-4	8.5E-4	9.7E-4
Radium-224	1.5E-3	1.3E-3	2.3E-3	1.1E-3	1.4E-3	1.1E-3	8.2E-4	1.9E-3	8.0E-4	1.2E-3	1.4E-3
Bismuth-214	2.1E-1	2.0E-1	2.3E-1	1.7E-1	2.0E-1	1.8E-1	1.7E-1	1.7E-1	1.7E-1	2.1E-1	2.1E-1
Bismuth-212	1.8E-2	1.7E-2	2.0E-2	1.5E-2	1.7E-2	1.5E-2	1.3E-2	1.7E-2	1.0E-2	1.9E-2	1.7E-2
Lead-214	3.6E-2	3.2E-2	4.9E-2	2.9E-2	3.6E-2	2.8E-2	2.3E-2	4.1E-2	1.6E-2	3.4E-2	3.4E-2
Lead-212	2.3E-2	2.0E-2	3.6E-2	1.6E-2	2.2E-2	1.7E-2	1.2E-2	2.8E-2	1.2E-2	1.8E-2	2.1E-2
Lead-210	2.6E-3	1.6E-3	4.5E-3	1.4E-3	2.2E-3	1.3E-3	9.7E-4	2.2E-3	1.4E-3	1.3E-3	2.2E-3
Polonium-210	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Thallium-208	4.2E-1	4.0E-1	4.5E-1	3.4E-1	4.0E-1	3.7E-1	3.4E-1	2.7E-1	3.8E-1	3.0E-1	4.2E-1

Table A-9. Dose conversion factors for use in preliminary
Clean Air Act assessments--continued

Radionuclide	Total body	Lung	Bone	Kidney	Thyroid	Liver	G. I. tract	Testis	Ovary	Spleen	Muscle
			Surface Exposure--rem-cm^2/microcurie-h (continued)								
Xenon-138	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Xenon-135m	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Xenon-135	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Xenon-133m	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Xenon-133	9.5E-3	7.3E-3	1.6E-2	6.6E-3	1.1E-2	6.3E-3	4.1E-3	9.7E-3	3.9E-3	5.2E-3	8.3E-3
Xenon-131m	1.0E-3	6.1E-4	1.5E-3	6.1E-4	1.0E-3	4.8E-4	2.4E-4	1.4E-3	4.1E-4	4.7E-4	9.3E-4
Cesium-137	8.2E-2	7.7E-2	1.0E-1	7.3E-2	8.6E-2	6.8E-2	5.9E-2	9.1E-2	3.0E-2	8.6E-2	8.2E-2
Cesium-134	2.2E-1	2.1E-1	2.6E-1	2.0E-1	2.0E-1	1.9E-1	1.5E-1	2.2E-1	1.1E-1	2.3E-1	2.2E-1
Iodine-133	1.0E-1	9.5E-2	1.2E-1	8.8E-2	1.0E-1	8.4E-2	7.3E-2	1.0E-1	4.7E-2	1.1E-1	1.0E-1
Iodine-131	5.4E-2	5.1E-2	6.7E-2	4.7E-2	5.7E-2	4.5E-2	3.8E-2	6.0E-2	2.0E-2	5.6E-2	5.4E-2
Iodine-129	6.8E-3	3.3E-3	9.9E-3	3.7E-3	6.9E-3	2.5E-3	7.8E-4	8.9E-3	2.3E-3	2.2E-3	6.0E-3
Iodine-125	1.3E-2	6.6E-3	2.0E-2	7.4E-3	1.4E-2	4.9E-3	1.6E-3	1.8E-2	4.7E-3	4.4E-3	1.2E-2
Technetium-99m	1.9E-2	1.6E-2	3.3E-2	1.4E-2	2.3E-2	1.4E-2	9.7E-3	1.7E-2	8.0E-3	1.1E-2	1.6E-2
Technetium-99	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Strontium-90	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Krypton-88	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Krypton-87	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Krypton-85m	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Krypton-85	3.0E-4	2.9E-4	3.7E-4	2.7E-4	3.2E-4	2.5E-4	2.2E-4	3.4E-4	1.1E-4	2.5E-4	1.1E-4
Krypton-83m	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Cobalt-60	3.5E-1	3.3E-1	3.7E-1	2.8E-1	3.1E-1	2.9E-1	2.7E-1	2.8E-1	2.8E-1	3.8E-1	3.5E-1
Cobalt-58	1.4E-1	1.3E-1	1.6E-1	1.3E-1	1.1E-1	1.2E-1	9.3E-2	1.3E-1	8.1E-2	1.4E-1	1.4E-1
Iron-55	2.1E-4	1.6E-5	3.1E-5	3.9E-6	7.8E-6	3.9E-6	7.8E-6	1.5E-4	7.8E-6	4.3E-5	7.0E-5
Argon-41	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Carbon-14	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Tritium	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0

Table A-10. Sources for dose conversion factors used in table A-9

Radionuclides	Inhalation	Ingestion	Air submersion and surface exposure
Americium-241	[a]EPA77	EPA77	Mo78
Plutonium-242,241,240,238	Mo78	Mo78	Mo78
Plutonium-239	[b]EPA77	[c]EPA77	Mo78
Uranium-238,235,234	Ho74	Mo78	Mo78
Uranium-236	ORNL78	ORNL78	Ki76
Uranium-233	Mo78	Mo78	Mo78
Protactinium-234m,234	Mo78	Mo78	Mo78
Thorium-234,232,231,228	Ho74	Mo78	Mo78
Thorium-230	Ho74	Ki76	Ki76
Radium-228,224	Ho74	Mo78	Ki76
Radium-226	[d]Ho74	[e]Wa78b	Ki76
Bismuth-214	-	-	Ki76
Bismuth-212	Ho74	Mo78	Mo78
Lead-214	-	-	Ki76
Lead-212	Ho74	Mo78	Mo78
Lead-210	Wa78a	[f]Wa78a	Mo78
Polonium-210	Wa78a	[g]Wa78a	Mo78
Thallium-208	-	-	Ki76
Xenon-138,135m,135,133m,133	Mo78	Mo78	Mo78
Xenon-131m	[h]Mo78	-	Ki76
Cesium-137,134	Mo78	Mo78	Mo78
Iodine-133,129	Mo78	Mo78	Mo78
Iodine-131,125	Ke76	Ke76	Ki76
Technetium-99m,99	Ki78	Ki78	Ki76
Strontium-90	Ki78	Ki78	Ki76

Table A-10. Sources for dose conversion factors used in table A-9--continued

Radionuclides	Inhalation	Ingestion	Air submersion and surface exposure
Krypton-88,87,85m,85,83m	Mo78	Mo78	Mo78
Cobalt-60,58	[i]Ki78	Mo78	Mo78
Iron-55	Mo78	Mo78	Mo78
Argon-41	Mo78	Mo78	Mo78
Carbon-14	Ki78	Ki78	Mo78
Tritium	Mo78	Mo78	Mo78

[a] Except total body (Mo78).
[b] Except total body, kidney, G.I. tract (Mo78).
[c] Except kidney, lung, and G.I. tract (Mo78).
[d] Except total body (El78).
[e] Except total body (El78) and G.I. tract (Ki76).
[f] Except lung and G.I. tract (Mo78).
[g] Except lung, G.I. tract and spleen (Mo78).
[h] Except lung (NRC77).
[i] Except spleen (Mo78).

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APPENDIX B
HEALTH RISK ASSESSMENT METHODOLOGY

The fatal cancer risks presented in this report were estimated from the risk/rem and risk/WL-year factors shown in table B-1. The risk/rem factors were developed from information in the BEIR report (BEIR72). The risk/WL-year conversion factors were developed primarily from information on uranium miner exposures (EPA79 Section 4.0).

Risk estimates are limited to fatal cancers only. Our current practice is to assume that for whole body exposure, the number of genetic health effects and the number of nonfatal cancers are each about the same as the number of fatal cancers (EPA77).

In applying these risk factors to the organ doses calculated by the AIRDOS-II code, the following modifications were necessary:

(1) Since AIRDOS-II calculates only the dose to bone (and not to bone marrow), a risk factor of 3×10^{-5} fatal cancers per person-rem was applied to the bone doses in order to take into consideration the risk to both the bone and red bone marrow. This factor is a composite of the risk factors in table B-1 of 4×10^{-5} for red bone marrow and 1×10^{-5} for bone (other organ) and was developed on the basis that the average ratio of red bone marrow dose to bone dose is about 0.5.

(2) A risk factor of 5×10^{-5} fatal cancers per person-rem was applied to soft tissue doses calculated by AIRDOS-II. This factor includes the risk to breast of 4×10^{-5} and the risk for one other soft tissue organ of 1×10^{-5}.

(3) The kidney and liver were used as the remaining two organs in the "all other" category.

(4) Since stomach doses were not calculated by AIRDOS-II, no corresponding risks were estimated.

(5) The total body risk factor was not used since dose estimates for specific organs were available.

The individual lifetime risks are the fatal cancers risks to individuals which would result from an average lifetime exposure (70 years) to the dose rates and working levels estimated for those individuals. The lifetime risk to the maximum individual was obtained by multiplying the dose equivalent rates and working level exposures by 70 to obtain the lifetime exposure and then multiplying this value by the risk/rem or risk/WL-year factors shown in table B-1.

The lifetime risk to the average individual was obtained by dividing the population exposures in person-rem/year and person-working levels by the total number of people in the exposed population to obtain average annual exposure rates and then proceeding as described above for the maximum individual.

The number of fatal cancers per year were obtained by multiplying the annual collective dose equivalents and working level exposures by the risk/rem or risk/WL-year factors in table B-1.

Table B-1. Risks of fatal cancer

Organ	Risk of fatal cancer (per person-rem)	(per person-WL-y)
Total body	2E-4	
Red bone marrow (leukemia)	4E-5	
Lung	4E-5	[a]2.1E-2
Breast (average for both sexes)	4E-5	
G. I. tract	2E-5	
Stomach	2E-5	
All others[b] (for each site)	1E-5	
Thyroid[c]	1E-6	

[a] Exposures from radon-222 decay products.
[b] Up to four sites other than those listed above.
[c] Nonfatal cancer risk of 1E-5.

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APPENDIX C
SOURCE TERM CALCULATIONS

C.1 Metal and Nonmetal Mining

C.1.1 Mines

Radon-222 emissions from mines were estimated either directly from measurement data on concentrations of radon-222 in mine ventilation air or in the absence of measurement data indirectly from the uranium-238 content of the ore.

Radon-222 emissions for iron, zinc, and clay mines were estimated from effluent measurement data as follows:

$$A = B \cdot C \cdot D \qquad (1)$$

where

A = radon-222 emissions, Ci/y

B = mine ventilation rate, ft^3/min

C = radon-222 concentration in the mine effluent, pCi/L

D = unit conversion factor = $1.49 \times 10^{-5} \; \dfrac{\text{L min Ci}}{\text{ft}^3 \text{ y pCi}}$

The values of the parameters A, B, and C are listed in table C.1-1.

Radon-222 emissions from mining operations are comprised of two components: radon-222 released during the actual mining of the ore and radon-222 released from ore surfaces exposed during the mining. Releases from copper, limestone, fluorspar, and bauxite mines, estimated from the uranium-238 content of the ore, were calculated as follows:

$$E = F \cdot G \cdot H \cdot I + Z \qquad (2)$$

where

E = radon-222 emissions, Ci/y

F = amount of ore mined annually, short ton/y or ST/y

G = concentration of uranium-238 in ore, ppm

H = fraction of radon released from ore = 0.2

I = unit conversion factor = $3.03 \times 10^{-7} \frac{\text{Ci}}{\text{ST ppm}}$

Z = annual release rate of radon resulting from the exposed ore body

Radioactive equilibrium was assumed between uranium and its daughters. The values of parameters E, F, G, Z are listed in table C.1-2.

Table C.1-1. Values of parameters A, B, C

Type of mine	A Radon emission rate (Ci/y)[a]	B Ventilation rate (ft^3/min)	C Radon effluent concentration (pCi/L)
Iron	67	500,000	9
Zinc	168	220,000	35
Clay	18	43,000	28

[a] These emission rates resulted from measurements performed by EPA during 1978 and 1979.

Table C.1-2. Values of parameters E, F, G, Z

Type of Mine	E Radon emission rate (Ci/y)	F Amount of ore mined (short ton/y)	G Uranium-238 concentration in ore (ppm)	Z Radon from exposed ore surface (Ci/y)
Copper	27 to 1518	10E+6	1 to 55[a]	27 to 1485[b]
Limestone	7	600,000	2[c]	7[d]
Fluorspar	0.2	15,000	2[e]	0.2[d]
Bauxite	10	200,000	8[f]	10[d]

[a] Source: Fi76.

[b] For an area of 2.6×10^6 m^2 (1 mile square) the rate of radon released from exposed ore surface is estimated to be 27 Ci/y per ppm of uranium-238.

[c] Source: EPA78. This value represents an average of 6 samples ranging from 0.93 to 1.92 ppm uranium-238 from the Calera mines in Alabama.

[d] The radon release rate from exposed ore surface is assumed to be one hundred times that resulting from mining.

[e] This value represents an average of 4 samples from the Cave in Rock mine in Illinois taken in 1978. The uranium-238 concentration ranged from 1.23 to 3.69 ppm.

[f] Source: Ad60.

C.1.2 Mills

The following formula was used to estimate the radioactive emissions at the various milling operations:

$$G = H \cdot I \cdot J \qquad (3)$$

where

G = uranium-238 release rate, µCi/y

H = uranium-238 concentration of ore, ppm

I = particulate emission rate, g/s

J = unit conversion factor = $7.2 \dfrac{\text{s } \mu\text{Ci}}{\text{y} \cdot \text{g} \cdot \text{ppm}}$

The values of G, H, I for the various milling activities are listed in table C.1-3.

Table C.1-3. Values of parameters G, H, I

Type of mill	G Uranium-238 release rate (Ci/y)	H Uranium-238 concentration in ore (ppm)	I Particulate emission rate (g/s)[a]
Iron	91	2	6.3
Copper	7.2 to 396	1 to 55	1
Zinc	3.6	3	0.16
Clay	6.0	6	0.14
Limestone	6.0	2	0.4
Fluorspar	1.7	2	0.12
Bauxite	9.0	8	0.15

[a] These emission rates are "rough estimates" based upon judgmental observations made during the EPA environmental mine sampling program. These values will be updated by actual measurements currently under way.

C.1.3 Tailings

The following formula was used to estimate the net radon-222 emissions from tailings piles:

$$K = (LM - NP)\,Q \qquad (4)$$

where

K = net radon-222 exhaled by the tailings pile, Ci/y

L = area of tailings pile in km^2

M = specific radon-222 exhalation rate of the tailings pile, pCi/m^2min

N = exhalation area without tailings pile, km^2

P = background exhalation rate = 30 pCi/m^2min

Q = unit conversion factor = $0.53 \frac{min\ \ Ci\ \ m^2}{y\ \ pCi\ \ km^2}$

The values of K, L, M, N are listed in table C.1-4.

Table C.1-4. Values of parameters K, L, M, N

	K	Dimensions of tailings pile[a]		L	M	N
Type of tailings	Radon-222 exhaled by tailings pile (Ci/y)	$A(km^2)$	$h(m)$	Radon-222 exhalation area of tailings pile (km^2)	Specific radon-222 exhalation rate of tailings pile[b] (pCi/m^2-min)	Radon-222 exhalation area without tailings pile (km^2)
Iron	160	5.2	12.2	5.3	87	5.2
Copper	0 to 1530	2.6	12.2	2.7	20 to 1100	2.6
Zinc	0	0.41	12.2	0.44	10[c]	0.41
Fluorspar	0	0.062	1.5	0.064	40	0.062
Bauxite	12	0.16	4.9	0.17	160	0.16

[a] Configuration of each tailings pile is assumed to be a parallelepiped with the area of base A and altitude h.

[b] These values were determined by substituting for H in the equation:

$$H \text{ ppm} \times 0.33 \text{ pCi/g}\cdot\text{ppm} \times \frac{60 \text{ pCi/m}^2\text{-min}}{\text{pCi/g}} = M, \text{ where H is the}$$

concentration of uranium-238 in the ore in which it is assumed that secular equilibrium exists between uranium-238 and its decay products. The radon-222 exhalation rate per pCi/g of uranium-238 used in this equation was calculated as described in Appendix C.2 using: $E = 0.2$, $D_e/v = .05\ cm^2/s$ and a density of $1.6\ g/cm^3$.

[c] In the case of zinc tailings, the measured emission rate was used.

C.2 Radon-222 Release Rates from Large Area Sources

Radon-222 releases from large area sources (e.g., ore or coal storage piles, waste piles, etc.) were estimated using a simple one-dimensional diffusion model (ORNL75) as follows:

$$J = C \cdot E\ (\lambda D_e/v)^{\frac{1}{2}} \qquad (5)$$

where

J = radon-222 flux from source surface, $\mathrm{pCi/cm^2\text{-}s}$

C = radium-226 concentration per unit volume in source material, $\mathrm{pCi/cm^3}$

E = radon emanation coefficient

λ = radon-222 decay constant = $2.1\mathrm{E}{-6}\ \mathrm{s^{-1}}$

D_e/v = diffusion coefficient/void fraction, $\mathrm{cm^2/s}$

The model assumes a source thick enough that no finite thickness correction is necessary.

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APPENDIX D
GLOSSARY OF TERMS AND ABBREVIATIONS

accelerator

A device for increasing the kinetic energy of charged elementary particles (for example, electrons or protons) through the application of electrical and/or magnetic forces.

activity

Radioactivity or radioactive materials.

activity

A measure of the rate at which a material is undergoing nuclear transformations. The special unit of activity used in this report is the curie (Ci).

AEC

Atomic Energy Commission (discontinued with formation of ERDA and NRC on January 19, 1975).

agreement states

Those states which, pursuant to Section 274 of the Atomic Energy Act of 1954, as amended, have entered into an agreement with the NRC for assumption of regulatory control of byproduct, source, and small quantities of special nuclear materials. Before approving such an agreement, NRC must determine that the State's radiation control program is compatible with NRC's regulatory program and is adequate to protect public health and safety.

alpha particle (α)

A positively charged particle emitted by certain radioactive materials. It is made up of two neutrons and two protons; hence, it is identical to the nucleus of a helium atom. The alpha particle has a nuclear mass number of four and charge of positive two.

alpha radiation

An emission of alpha particles (helium nuclei) from a material undergoing nuclear transformation.

aquifer

A water-bearing layer of permeable rock or soil.
A subsurface formation containing sufficient saturated permeable material to yield significant quantities of water.

atomic number

The number of protons in the nucleus of a chemical element.

background radiation

Radiation in the natural human environment originating from cosmic rays and from the naturally radioactive elements of the earth, including those within the human body. The level of radioactivity in an area which is produced by sources other than the one of specific interest.

beta particle (B)

An elementary particle emitted from a nucleus during radioactive decay. It has a single negative electric charge and a mass equal to 1/1837 that of a proton. A beta particle is identical to an electron.

biota

Plant and animal life.

body burden

The amount of a specified radioactive material or the summation of the amounts of various radioactive materials present in an animal or human body at a specified time.

boiling water reactor (BWR)

A type of nuclear power reactor that employs ordinary water (H_2O) as coolant and moderator and allows bulk boiling in the core so that steam is generated in the primary reactor vessel.

burial grounds

Areas designated for storage of containers of radioactive wastes by near-surface burial in geologic media.

byproduct material

Radioactive material produced in a nuclear reactor, ancillary to the reactors main purpose of producing power or fissile materials. Fission products are usually considered to be byproduct material.

calcination

Solidification method of disposal of liquid wastes involving atomizing and coating of liquid on small granular solids, followed by heating to drive off moisture.

calcine

Material heated to a temperature below its melting point to bring about loss of moisture and oxidation resulting in a chemically stable form.

canyon building

A heavily shielded building used in the chemical processing of irradiated fuel and target elements. Operation and maintenance is by remote control.

Ci

curies

clay

As a soil separate, clay contains mineral soil particles that are less than 0.002 millimeters in diameter. As a soil textural class, the soil material is 40 percent or more clay, less than 45 percent sand, and less than 40 percent silt.

cladding

The outer jacket of nuclear fuel elements preventing corrosion of the fuel and the release of fission products into the coolant. Stainless steel and zirconium alloys are common cladding materials.
Synonym: hull

coarse fragments

Gravel, cobblestones, or stones in soil that range in size from 2 millimeters to 3 feet.

controlled area

Any specific region of the Hanford Reservation into which entry by personnel is regulated by physical barrier or procedure.

counts per minute (cpm)

The number of events per unit time recorded by an instrument designed to detect radioactive particles; especially used to indicate the relative amount of radioactive contamination.

criticality safety

Those procedures and understandings necessary to the handling of fissile materials in a manner that will prevent them from reaching a critical condition.

curie

The basic unit used to describe the intensity of radioactivity in a sample of material. One curie (Ci) equals 37 billion disintegrations per second.

daughter products

The nuclides formed by the radioactive disintegration of a first nuclide (parent).

deactivated

The condition of a facility or disposal site where steps have been taken to preclude further operation or the further addition of waste materials.

decay

The spontaneous radioactive transformation of one nuclide into a different nuclide or into a different energy state of the same nuclide. Every decay process has a definite half-life.

decay chain

The sequence of radioactive disintegrations in succession from one nuclide to another until a stable daughter is reached.

DOE

Department of Energy. Established by Executive Order in October 1977. Comprises the following former agencies: Energy Research and Development Administration, Federal Energy Administration, Federal Power Commission, and parts of the Department of Interior.

dose

The energy imparted to matter by ionizing radiation per unit mass of irradiated material at a specific location. The unit of absorbed dose is the rad. A general term indicating the amount of energy absorbed from incident radiation by a specified mass.

dose commitment

The integrated dose which results from an intake of radioactive material when the dose is evaluated from the beginning of intake to a later time (usually 50 years;) also used for the long term integrated dose to which people are considered committed because radioactive material has been released to the environment.

environmental surveillance

A program to monitor the impact on the surrounding region of the discharges from industrial operations.

enriched uranium

Uranium in which the percentage of the fissionable isotope uranium-235 has been increased above the 0.7% contained in natural uranium.

EPA

Environmental Protection Agency

ERDA

Energy Research and Development Administration (the nuclear program components of ERDA were formerly part of the AEC), now part of Department of Energy.

exposure

The condition of being made subject to the action of radiation.

extraction

A chemical process for selectively removing materials from solutions.

fallout

Those radioactive materials deposited on the earth's surface and in the atmosphere following the detonation of nuclear weapons.

fertile material

A material (for example, uranium-238) not fissionable, but which can be converted into a fissionable material by irradiation in a reactor.

fission

The splitting of a heavy nucleus into two roughly equal parts (which are nuclei of lighter elements), accompanied by the release of a relatively large amount of energy and frequently one or more neutrons.

fission products

Nuclei formed by the fission of heavy elements. Many are radioactive. Examples: strontium-90, cesium-137.

fissionable material

Any material readily fissioned by neutrons, for example, uranium-235 and plutonium-239.

fuel cycle (nuclear, reactor)

The series of steps involved in supplying fuel for nuclear power reactors. It includes mining, refining, the original fabrication of fuel elements, their use in a reactor, chemical processing to recover the fissionable material remaining in the spent fuel, re-enrichment of the fuel material, and refabrication into new fuel elements.

fuel (nuclear, reactor)

Fissionable material used as the source of power when placed in a critical arrangement in a nuclear reactor.

fuel separation
(fuel reprocessing)

Processing of irradiated (spent) nuclear reactor fuel to recover useful materials as separate products, usually involving separation into plutonium, uranium, and fission products.

fuel rod

A tube containing UO_2 or mixed oxide fuel; part of a fuel assembly in a nuclear reactor.

fuel element

A tube, rod, or other form into which fissionable material is fabricated for use in a reactor.

g

grams

gamma rays (γ)

High-energy, short-wavelength electromagnetic radiation emitted by a nucleus. Gamma radiation usually accompanies alpha and beta emissions and always accompanies fission.

gastrointestinal dose (GI dose)

The dose to the stomach and lower tract of humans and animals via external exposure or via internal transport of radioactive material.

g

grams

ground water

Water in the zone of saturation beneath the land surface.

half-life

The time in which half the atoms of a given quantity of a particular radioactive substance disintegrate to another nuclear form. Measured half-lives vary from millionths of a second to billions of years.

half-life, biological

The time required for a living organism to eliminate, by natural processes, half the amount of a substance that has entered it.

half-life, effective

The time required for a radionuclide contained in a biological system to reduce its activity by half due to the combined result of radioactive decay and biological elimination.

heavy water

Deuterium oxide, D_2O. Water in which normal hydrogen atoms have been replaced with deuterium atoms. Having a low neutron absorption cross section, D_2O readily dissipates the energy of the high-energy neutrons which sustain a fission reaction. Hence, D_2O is used as a moderator in some nuclear reactors. In SRP reactors it is used as both the moderator and the primary coolant.

HEPA

High efficiency particulate air filter. A type of filter designed to remove 99.9 percent of particles down to 0.3 μm in diameter from a flowing air stream.

hood

A canopy and exhaust duct used to confine hazardous materials in order to reduce the exposure of industrial workers.

hypothetical maximum individual (max man)

A postulated person who is assumed to receive the maximum credible radiological dose through each of the exposure pathways from the source being considered.

inactive

The condition of a facility or disposal site which is not presently being operated or to which materials are not being added.

ICRP

International Commission on Radiological Protection

ion exchange

A reversible chemical reaction between a solid and a fluid mixture by means of which ions may be interchanged.

isotope

One of two or more forms of an element that differ in atomic weight. Nuclides with the same atomic number, (i.e., the same chemical element, characterized by the number of protons contained in the atomic nucleus) but with different atomic masses (i.e., different numbers of neutrons contained in the nucleus). Although chemical properties are the same, radioactive and nuclear (radioactive decay) properties may be quite different for each isotope of an element.

km

kilometers (1 kilometer = 1000 meters or 0.621 mile)

leaching

Extracting material from a solid by passing water or a solution through the solid material.

light water

Normal water (H_2O), as distinguished from heavy water (D_2O).

light water reactor

A reactor in which ordinary water (H_2O) is used as the coolant and moderator. In such reactors the water is either allowed to boil (boiling water reactor or BWR) or pressurized to prevent boiling (pressurized water reactor or PWR).

long-lived nuclides

Radioactive isotopes with half-lives greater than about 30 years. Most long-lived nuclides of interest to waste management have half-lives on the order of thousands to millions of years. For example:

^{239}Pu - 24,400 years; ^{99}Tc - 2.1 x 10^5 years;
^{129}I - 1.6 x 10^7 years.

low-level waste

Wastes containing types and concentrations of radioactivity such that shielding to prevent personnel exposure is not required.

m

1. meter
2. as prefix, milli. See "milli."

micro (μ)

Prefix indicating one millionth (1 microgram = 1/1,000,000 of a gram or 10^{-6} gram).

milli

Prefix indicating one thousandth

millirem

One thousandth of a rem

ml

milliliters

moderator

A material, such as heavy water, used in a reactor to slow down the high-velocity neutrons which sustain a fission reaction.

mR

millirads

mrem

millirem

nano

Prefix indicating one thousandth of a micro unit (1 nanocurie = 1/1000 of a microcurie or 10^{-9} curie).

natural (normal) uranium

Uranium as found in nature. It is a mixture of the fertile uranium-238 isotope (99.3%), the fissionable uranium-235 isotope (0.7%), and a minute percentage of uranium-234.

neutron

An uncharged elementary particle with a mass nearly equaled by that of the proton. Neutrons are part of the fission chain reaction in a nuclear reactor. They can also be generated by spontaneous fission and by collision of high energy γ rays and α particles with some nuclei.

NRC

Nuclear Regulatory Commission (formerly part of AEC).

nuclide

Any atomic nucleus specified by its atomic weight, atomic number, and energy state. A radionuclide is a radioactive nuclide.

nuclear radiation

Particles and electromagnetic energy given off by transformations occurring in the nucleus of an atom.

off-gas

The gas given off in any stage of an industrial process.

pCi

picocuries

permeability, soil

That quality of the soil that enables it to transmit water or air. Terms used to describe permeability in inches per hour are:

Very slow	Less than 0.06 inches
Slow	0.6 to 0.2 inches
Moderately slow	0.2 to 0.6 inches
Moderate	0.6 to 2.0 inches
Moderately rapid	2.0 to 6.0 inches
Rapid	6.0 to 2.0 inches
Very rapid	More than 20 inches

person-rem

Used as a unit of population dose; the average dose per individual expressed in rems times the population affected.

pH

A measure of the hydrogen ion concentration in aqueous solutions. Acidic solutions have a pH from zero to 7. Basic solutions have a pH from 7 to 14.

pico

Prefix indicating one millionth of a micro unit (1 picocurie = 1/1,000,000 of a microcurie or 10^{-12} curie).

population dose
(population exposure)

The summation of individual radiation doses received by all those exposed to the source or event being considered.

plutonium

A radioactive element with atomic number 94. Its most important isotope is fissionable plutonium-239, produced by neutron irradiation of uranium-238.

power reactor

A nuclear reactor designed to produce heat for conversion to electrical energy or mechanical propulsion.

ppm

parts per million

precipitation scavenging

The process by which rain or snow removes particulates or reactive vapors from the atmosphere and deposits them on the ground surface.

production reactor

A nuclear reactor designed primarily for large-scale production of plutonium, tritium, and other radionuclides by neutron irradiation. A nuclear reactor designed for transforming one nuclide into another; usually, a conversion of natural uranium into plutonium.

Purex

A solvent extraction process in which uranium and plutonium are selectively separated from each other and from fission products by extraction from nitric acid solutions with tributylphosphate in a hydrocarbon diluent.

rad

Radiation absorbed dose. The basic unit of absorbed dose of ionizing radiation. One rad is equal to the absorption of 100 ergs of radiation energy per gram of matter.

radiation (ionizing)

Particles and electromagnetic energy emitted by nuclear transformations which are capable of producing ions when interacting with matter; gamma rays and alpha and beta particles are primary examples in INEL waste.

radioiodines

Isotopes of iodine which are radioactive.

radioactive (decay)

Property of undergoing spontaneous nuclear transformation in which nuclear particles or electromagnetic energy are emitted.

radioactivity

The spontaneous decay or disintegration of unstable atomic nuclei, accompanied by the emission of radiation.

radionuclide

An unstable nuclide of an element that decays or disintegrates spontaneously, emitting radiation.

radioisotope

A radioactive isotope. An unstable isotope of an element that decays or disintegrates spontaneously, emitting radiation. More than 1300 natural and artificial radioisotopes have been identified.

radwaste

Waste materials which are radioactive.

reactor

A device by means of which a fission chain reaction can be initiated, maintained, and controlled. A nuclear reactor.

recycle

The returning of uranium and plutonium (recovered in spent fuel reprocessing) for reuse in new reactor fuel elements.

release limit
(release guide)

A control number which regulates the concentration or amount of radioactive material released to the environment in an industrial situation; usually dose to persons in the environment derived from environmental behavior of the released material so that the dose is kept below a selected control value.

rem

A dose unit which takes into account the relative biological effectiveness (RBE) of the radiation. The rem ("roentgen equivalent man") is defined as the dose of a particular type of radiation required to produce the same biological effect as one roentgen of (0.25 Mev) gamma radiation. A millirem (mrem) is one thousandth of a rem.

roentgen (R)

A measure of the ability of gamma or X rays to produce ionization in air. One roentgen corresponds to the absorption of about 86 ergs (100 ergs = 6.24×10^7 million electron volts, Mev) of energy from X- or gamma radiation, per gram of air. The corresponding absorption of energy in tissue may be from one-half to two times as great, depending on the energy of the radiation and the chemical composition of the tissue. The roentgen is thus more useful as a measure of the amount of gamma or X rays to which one is exposed than as a measure of the dose of such radiation actually received.

reprocessing

Chemical processing of irradiated nuclear reactor fuels to remove desired constituents.

retention basin

An excavated and lined area used to hold contaminated fluids until radioactive decay reduces activities to levels permissible for release.

retired facility

A facility which has been shut down with no intentions of restarting and which has had appropriate controls and safeguards placed on it.

salt cake

The solid residue resulting from a concentration of high-level liquid waste in underground waste storage tanks.

scavenging

See precipitation scavenging.

separations

Chemical processes used to separate nuclear products from byproducts and from each other.

short-lived nuclides

Radioactive isotopes with half-lives not greater than about 30 years, e.g., ^{137}Cs and ^{90}Sr.

solidification

Conversion of radioactive waste to a dry, stable solid.

solid wastes (radioactive)

Either solid radioactive material or solid objects which contain radioactive material or bear radioactive surface contamination.

solvent extraction

A process in which materials are selectively removed from an aqueous solution by contact with an immiscible organic solvent.

source material

Uranium or thorium or any ores which contain at least 0.05% of uranium or thorium

source term

Release rates (in curies per year) to the atmosphere from each point or area source are known collectively as the source term.

special nuclear material (SNM)

Plutonium, ^{233}U, ^{235}U, or uranium enriched to a higher percentage than normal of the 233 or 235 isotopes.

stability (atmospheric)

A description of the atmospheric forces on a parcel of air following vertical displacement in an atmosphere otherwise in hydrostatic equilibrium; if the forces tend to return the parcel to its original level, the atmosphere is stable; if the forces tend to move the parcel further in the direction of displacement, the atmosphere is unstable, and if the air parcel tends to remain at its new level the atmosphere has neutral stability.

standby

The condition where a facility or burial ground, etc., is placed in a nonoperating condition but is maintained in readiness for subsequent operation

storage basin

A water-filled facility for holding irradiated reactor fuels with the water acting as a shield.

tank

A large metal container located underground for storage of liquid wastes

tank farm

An installation of interconnected underground containers (tanks) for storage of high-level waste.

thorium

A naturally radioactive element with atomic number 90 and, as found in nature, and atomic weight of approximately 232. The fertile thorium 232 isotope is abundant and can be transmuted to fissile uranium 233 by neutron irradiation.

total body dose

The radiation dose to the entire body.

tracer

A radionuclide(s) or chemical introduced in minute quantities to a system or process for the purpose of using radiation or chemical detection techniques to follow the behavior of the process or system.

transmutation

The process whereby one nuclide changes (or is changed) into another, usually by means of bombardment with nuclear particles.

transuranic elements

Elements with mass numbers greater than 92, including neptunium, plutonium, americium, and curium.

trench

A long and narrow excavation in the ground for solid waste. Unless qualifying descriptions are given, a trench is unlined, and its walls are unsupported. After the solid wastes are placed in position, the trench is filled to grade level with some of the removed soil.

TRIGA reactor

A research and training pool type nuclear reactor built by Gulf General Atomics which has a compact zirconium hydride core and which can operate in either a steady state or a pulsing mode.

tritium

A radioactive isotope of hydrogen with two neutrons and one proton in the nucleus. It is heavier than deuterium (heavy hydrogen). Tritium (T or ^{3}H) is used in industrial thickness gages, as a label in tracer experiments, in controlled nuclear fusion experiments, and in thermonuclear weapons. It is produced primarily by neutron irradiation of lithium-6.

water table

Upper boundary of an unconfined aquifer surface below which saturated groundwater occurs; defined by the levels at which water stands in wells that barely penetrate the aquifer.

wind rose

A diagram designed to show the distribution of prevailing wind directions at a given location; some variations include wind speed groupings by direction.

uranium

A naturally radioactive element with the atomic number 92 and an atomic weight of approximately 238. The two principal naturally occurring isotopes are the fissionable uranium-235 (0.7% of natural uranium) and the fertile uranium-238 (99.3% of natural uranium).

USAEC

United States Atomic Energy Commission (see AEC)

waste, radioactive

Equipment and materials (from nuclear operations) that are radioactive or have radioactive contamination and for which there is not recognized use or for which recovery is impractical.

water table

The upper surface of the ground water.

μ

mu, a prefix. Same as "micro."

μCi

microcuries

μg

micrograms

μm

micrometers

D-20

APPENDIX E
LIST OF ELEMENTS

ELEMENT	SYMBOL	ATOMIC NUMBER
actinium	Ac	89
aluminum	Al	13
americium	Am	95
antimony	Sb	51
argon	Ar	18
arsenic	As	33
astatine	At	85
barium	Ba	56
berkelium	Bk	97
beryllium	Be	4
bismuth	Bi	83
boron	B	5
bromine	Br	35
cadmium	Cd	48
calcium	Ca	20
californium	Cf	98
carbon	C	6
cerium	Ce	58
cesium	Cs	55
chlorine	Cl	17
chromium	Cr	14
cobalt	Co	27
columbium	Nb	(see niobium)
copper	Cu	29
curium	Cm	96
dysprosium	Dy	16
einsteinium	Es	99
erbium	Er	68
europium	Eu	63
fermium	Fm	100
fluorine	F	9
francium	Fr	87
gadolinium	Gd	64
gallium	Ga	31
germanium	Ge	32
gold	Au	79
hafnium	Hf	72
helium	He	2
holmium	Ho	67
hydrogen	H	1
indium	In	49
iodine	I	53
iridium	Ir	77
iron	Fe	26
krypton	Kr	36
lanthanum	La	57
lawrencium	Lw	103
lead	Pb	82
lithium	Li	3
lutetium	Lu	71
magnesium	Mg	12
manganese	Mn	25
mendelevium	Md	101
mercury	Hg	80
molybdenum	Mo	42
neodymium	Nd	60
neon	Ne	10
neptunium	Np	93
nickel	Ni	28
niobium	Nb	41
nitrogen	N	7
nobelium	No	102
osmium	Os	76
oxygen	O	8
palladium	Pd	46
phosphorus	P	15
platinum	Pt	78
plutonium	Pu	94
polonium	Po	84
potassium	K	19
praseodymium	Pr	59
promethium	Pm	61
protactinium	Pa	91
radium	Ra	88
radon	Rn	86
rhenium	Re	75
rhodium	Rh	45
rubidium	Rb	37
ruthenium	Ru	44
samarium	Sm	62
scandium	Sc	21
selenium	Se	34
silicon	Si	14
silver	Ag	47
sodium	Na	11
strontium	Sr	38
sulfur	S	16
tantalum	Ta	73
technetium	Tc	43
tellurium	Te	52
terbium	Tb	65
thallium	Tl	81
thorium	Th	90
thulium	Tm	69
tin	Sn	50
titanium	Ti	22
tungsten	W	74
uranium	U	92
vanadium	V	23
wolfram	W	(see tungsten)
xenon	Xe	54
ytterbium	Yb	70
yttrium	Y	39
zinc	Zn	30
zirconium	Zr	40

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